PART III.—Complete This Part Only if the Request is for Approval of a Collection of Information Under the Paperwork Reduction Act and 5 CFR 1320.

13. Abstract—Describe needs, uses and affected public in 50 words or less "Nuclear Facilities, Nuclear Power Plant Construction"

10 CFR 50 specifies technical information and data to be retained as records or submitted to the NRC by applicants and licensees for construction and/or operation of nuclear power plants.

14. Type of information collection *(check only one)*

Information collections not contained in rules

1 ☐ Regular submission
2 ☐ Emergency submission *(certification attached)*

Information collections contained in rules

3 ☒ Existing regulation *(no change proposed)*
4 ☐ Notice of proposed rulemaking (NPRM)
5 ☐ Final, NPRM was previously published
6 Final or interim final without prior NPRM
A ☐ Regular submission
B ☐ Emergency submission *(certification attached)*
7. Enter date of expected or actual Federal Register publication at this stage of rulemaking *(month, day, year)* _____

15. Type of review requested *(check only one)*

1 ☐ New collection
2 ☒ Revision of a currently approved collection
3 ☐ Extension of the expiration date of a currently approved collection without any change in the substance or in the method of collection
4 ☐ Reinstatement of a previously approved collection for which approval has expired
5 ☐ Existing collection in use without an OMB control number

16. Agency report form number(s) *(include standard/optional form number(s))*

N/A

17. Annual reporting or disclosure burden *

1 Number of respondents	167
2 Number of responses per respondent	27
3 Total annual responses *(line 1 times line 2)*	4,549
4 Hours per response	809.2
5 Total hours *(line 3 times line 4)*	3,681,009

18. Annual recordkeeping burden *

1 Number of recordkeepers	167
2 Annual hours per recordkeeper	7,922.5
3 Total recordkeeping hours *(line 1 times line 2)*	1,323,064
4 Recordkeeping retention period	Life years

19. Total annual burden

1 Requested *(line 17-5 plus line 18-3)*	4,997,109
2 In current OMB inventory	3,701,859
3 Difference *(line 1 less line 2)*	+1,295,250
Explanation of difference	
4 Program change	+36,400
5 Adjustment	+1,258,850

20. Current *(most recent)* OMB control number or comment number

3150-0011

21. Requested expiration date

3 years from approval date

22. Purpose of information collection *(check as many as apply)*

1 ☐ Application for benefits
2 ☐ Program evaluation
3 ☐ General purpose statistics
4 ☒ Regulatory or compliance
5 ☐ Program planning or management
6 ☐ Research
7 ☐ Audit

23. Frequency of recordkeeping or reporting *(check all that apply)*

1 ☒ Recordkeeping

Reporting

2 ☒ On occasion
3 ☐ Weekly
4 ☒ Monthly
5 ☐ Quarterly
6 ☒ Semi-annually
7 ☒ Annually
8 ☐ Biennially
9 ☐ Other *(describe)* _____

24. Respondents' obligation to comply *(check the strongest obligation that applies)*

1 ☐ Voluntary
2 ☐ Required to obtain or retain a benefit
3 ☒ Mandatory

25. Are the respondents primarily educational agencies or institutions or is the primary purpose of the collection related to Federal education programs? ☐ Yes ☒ No

26. Does the agency use sampling to select respondents or does the agency recommend or prescribe the use of sampling or statistical analysis by respondents? ☐ Yes ☒ No

27. Regulatory authority for the information collection

10 CFR 50 , or _____ FR _____ ; or, Other *(specify)* _____

Paperwork Certification

In submitting this request for OMB approval, the agency head, the senior official or an authorized representative, certifies that the requirements of 5 CFR 1320, the Privacy Act, statistical standards or directives, and any other applicable information policy directives have been complied with.

Signature of program official	Date

Signature of agency head, the senior official or an authorized representative	Date
Patricia G. Norry, DSO for Information Resources Management	5-6-91

* Numbers are rounded

9105130262 910506
PDR ORG EUSOMB
PDR

PDR

Standard Form 83
(Rev. September 1983)

Request for OMB Review

DESIGNATED ORIGINAL

Charlotte Smith

Important

Read instructions before completing form. Do not use the same SF 83 to request both an Executive Order 12291 review and approval under the Paperwork Reduction Act.

Answer all questions in Part I. If this request is for review under E.O. 12291, complete Part II and sign the regulatory certification. If the request is for approval under the Paperwork Reduction Act and 5 CFR 1320, skip Part II, complete Part III and sign the paperwork certification.

Send three copies of this form, the material to be reviewed, and for paperwork—three copies of the supporting statement, to:

Office of Information and Regulatory Affairs
Office of Management and Budget
Attention: Docket Library, Room 3201
Washington, DC 20503

PART I.—Complete This Part for All Requests.

1. Department, agency and Bureau/office originating request

U.S. Nuclear Regulatory Commission

2. Agency Code

3 1 5 0

3. Name of person who can best answer questions regarding this request

Hazel Smith

Telephone Number

301 492-0075

4. Title of information collection or rulemaking

10 CFR 50, Domestic Licensing of Production and Utilization Facilities

5. Law authority for information collection or rulemaking (Title, U.S.C., Public Law, or Executive Order)

42 USC 2201(o)

6. Affected public (check all that apply)

a. ☐ Individuals or households
b. ☐ State or local governments
c. X Businesses (including farms)
d. ☐ Federal agencies or employees
e. X Non-profit institutions

PART II.—Complete This Part Only if the Request is for OMB Review Under Executive Order 12291

7. Regulation identifier Number (RIN)

8. Type of Submission (check appropriate categories)

Classification:
1. ☐ Major
2. ☐ Nonmajor

Stage of development:
☐ Proposed
☐ Final
☐ Interim final
☐ Final without prior proposal

Type of review requested:
1. ☐ Standard
2. ☐ Pending
3. ☐ Emergency
4. ☐ Statutory or judicial deadline

9. CFR section affected

CFR

10. Does this regulation contain reporting or recordkeeping requirements that require approval under the Paperwork Reduction Act and 5 CFR 1320?

11. If a major rule, is there a regulatory impact analysis attached?

If "No," did OMB waive the analysis?

Certification for Regulatory Submissions

In submitting this request for OMB review, the agency head (or his delegate of not below the level of program official) certifies that the requirements of E.O. 12291 and applicable policy directives have been complied with.

Signature of program official

Date

Signature of authorized regulatory contact

Date

12. (OMB use only)

Previous editions obsolete
NSN 7540-01-134-4034

Standard Form 83 (Rev. 9-83)

1

DF07

Standard Form 83
Rev. September 1983

Request for OMB Review

Important

Read instructions before completing form. Do not use the same SF 83 to request both an Executive Order 12291 review and approval under the Paperwork Reduction Act.

Answer all questions in Part I. If this request is for review under E.O. 12291, complete Part II and sign the regulatory certification. If the request is for approval under the Paperwork Reduction Act and 5 CFR 1320, skip Part II, complete Part III and sign the paperwork certification.

Send three copies of this form, the material to be reviewed, and (for paperwork—three copies of the supporting statement, to:

Office of Information and Regulatory Affairs
Office of Management and Budget
Attention: Docket Library, Room 3201
Washington, DC 20503

PART I.—Complete This Part for All Requests.

1. Department, agency and Bureau, office originating request

U.S. Nuclear Regulatory Commission

2. [illegible]

3 1 5 0

3. Name of person who can best answer questions regarding this request

Hazel Smith

Telephone [illegible]

301 492-0075

4. Title of information collection or rulemaking

10 CFR 50, Domestic Licensing of Production and Utilization Facilities

5. Legal authority for information collection or rulemaking (Statute and section number)

42 USC 2201(o)

6. Affected public (check all that apply)

1. ☐ Individuals or households
2. ☐ Farms
3. ☐ Business or other for-profit
4. ☐ Federal agencies or employees
5. ☐ State or local governments
6. X Small businesses or organizations

X

X

PART II.—Complete This Part Only if the Request is for OMB Review Under Executive Order 12291

7. Regulation identifier Number (RIN)

8. Type of submission (check one in each category)

Classification

1. ☐ Major
2. ☐ Nonmajor

Stage of development

1. ☐ Proposal (NPRM)
2. ☐ Final, with [illegible]
3. ☐ Final, without [illegible]

Type of review requested

1. ☐ Statutory
2. ☐ Court ordered
3. ☐ Agency initiated
4. ☐ Statutory/court ordered deadline

9. CFR section affected

[illegible] CFR [illegible]

10. Does this regulation contain reporting or recordkeeping requirements that require OMB approval under the Paperwork Reduction Act and 5 CFR 1320?

11. If a major rule, is there a regulatory impact analysis attached?

If "No," did OMB waive the analysis

Certification for Regulatory Submissions

In submitting this request for OMB review, the authorized regulatory contact and the program official certify that the requirements of E.O. 12291 and any applicable policy directives have been complied with.

Signature of program official

Date

1/1

Signature of authorized regulatory contact

Date

12. *(OMB use only)*

Previous editions obsolete
NSN 7540-00-634-4034

Standard Form 83 [illegible]
Prescribed by [illegible]
5 CFR 1320.9(a)(1)

DF02

OMB Supporting Statement for 10 CFR Part 50

Domestic Licensing of Production and Utilization Facilities

(OMB Clearance No. 3150-0011)

GENERAL* DESCRIPTION OF THE INFORMATION COLLECTION

The 10 CFR Part 50 regulations "Domestic Licensing of Production and Utilization Facilities, are promulgated by the U.S. Nuclear Regulatory Commission (NRC) pursuant to the Atomic Energy Act of 1954, as amended (the Act), to provide for the licensing and regulation of production and utilization facilities. They contain the reporting, recordkeeping and application requirements as they are generally applied in the NRC's licensing and regulatory process. Specific requirements for each licensee are contained in documents called "Technical Specifications" that are issued for every utilization facility licensed to operate. (See 10 CFR 50.36 and Part 2 of this submittal.) Guidance on acceptable means of complying with 10 CFR 50 is provided through publications called NRC "Regulatory Guides." These guides often cite standards and other requirements established by national standards bodies such as the American National Standards Institute (ANSI) and the American Society of Mechanical Engineers (ASME).

The provisions encompassed within 10 CFR 50 affect various types of facilities, including nuclear power plants, research reactors and test reactors, at various stages in the licensing process, including application, construction, operation, amendment, suspension, renewal and shutdown. Therefore, the number of respondents actually affected by each requirement varies depending on the number of licensing requests initiated and/or completed and the number of regulatory reports required by operating events and/or conditions.

Reporting requirements obviously are directed toward licensees or applicants. However, reporting requirements may not be reactor specific, but they may be of a type that applies to a site** which is occupied by one or more reactors that have different licensees. Other requirements may be utility specific and, thus, refer to several reactors at more than one site. These considerations may cause apparent conflicts in the use of the terms licensees, reactor sites, facilities, or plants in our individual estimates of burden.

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*Specific discussion pertinent to the various sections of Part 50 are included in Parts 1 - 26 enclosed with this transmittal portion of the 10 CFR Part 50 Supporting Statement.

**There are currently 74 nuclear power reactor sites.

For estimating purposes, NRC has assumed the following annual average number of respondents:

Operating Power Reactor Licenses - 111
Construction Permits for Power Reactors - 8
New Applications for Construction Permits - 0
New Applications for Operating Licenses - 0
New Applications for Standard Designs - 0
Applications for License Amendments - 1100
Critical Facilities and Research/Test Reactors - 48
Sites for Nuclear Power Plants - 74

Recordkeeping Requirements

Recordkeeping requirements by 10 CFR Part 50 are of two broad types. The first type is the simple maintenance of copies of reports, letters, and other written documentation that already exist because of a reporting requirement found elsewhere in the regulations or in the license and technical specifications. The second type of recordkeeping is the generation and storage of reports because the information in the reports may need to be referred to for assessments or subsequent evaluation of occurrences at the facility.

The large volume of records which is kept for 10 CFR Part 50 is required primarily by the technical specifications, the quality assurance program, reports of changes specified in 50.59(b) and environmental qualification of equipment. Thus, a specific recordkeeping burden has been calculated for each of these technical areas. For all other technical areas, the recordkeeping burden was estimated to be 10 percent of the total burden (recordkeeping plus reporting).

Record Retention Periods

The NRC codified a recordkeeping rule that established four basic retention periods for all records that must be retained to meet the recordkeeping requirements the NRC imposes on its licensees and applicants. All proposed rules containing recordkeeping requirements must specify one of these four standardized retention periods. Further, the agency has established a policy that all information collection requirements imposed upon licensees and applicants must be contained in its regulations. Therefore, the NRC's technical, licensing, generic requirements, and information resources management staffs will carefully scrutinize guidance documents to identify information collections that are being imposed on the licensees to determine if they are mandatory or volunatry. Where appropriate, procedures will be implemented to ensure that the data required to be submitted or retained is clear to the licensees and applicants.

Additional Requirements

This submittal incorporates finalized and proposed information collection requirements under 10 CFR 50 that have been approved by OMB since our last overall submittal for 10 CFR Part 50 dated December 23, 1986. These

requirements are itemized below and are described in detail in the cognizant supporting statements.

Final New Rules

1. Station Blackout - Section 50.63

2. Decommissioning Rule, Sections 50.33(k)(1) and (2); 50.75

3. Bankruptcy Notification Requirements, Section 50.54(cc)

4. Licensee Notification to NRC, Section 50.74 which was issued in conjunction with the revised 10 CFR 55 (See OMB Clearance No. 3150-0018).

Final Amended Rules

1. Acceptance Criteria for Emergency Core Cooling, Section 50.46 and Appendix K

2. Records Retention Rule, Multiple Sections

3. Property Damage Insurance, Section 50.54(w)

4. Pressurized Thermal Shock Events, Section 50.61

Proposed New Rule

1. Maintenance Programs for Nuclear Power Plants, Section 50.65

Proposed Amended Rules

1. Emergency Response Data Systems, Section 50.72(a)(4)

2. Containment Leakage, Appendix J

3. ASME Codes, Section 50.55a

4. Design Deficiencies, Section 50.55(e)

5. Hydrogen Control, Section 50.44(c)

6. Standard Liquid Control System - Flow Control Conditions, Section 50.62, ATWS

This submittal does not address the information collection requirements specified in 10 CFR 50.73, "Licensee Event Reporting System." The burden associated with this regulation is encompassed within OMB Clearance No. 3150-0104.

In submitting this request for approval of a revision to the OMB clearance for 10 CFR 50, the NRC realizes its importance and complexity are such that our staff must work closely with yours. Ms. Brenda Jo Shelton (301-492-8132) of our Information and Records Management Branch is available to arrange for the participation of any NRC staff or legal representative if needed by OMB.

A. JUSTIFICATION

1. Need for the Collection of Information

The NRC is authorized by Congress to have responsibility and authority for the licensing and regulation of nuclear power plants, research and test facilities, fuel reprocessing plants and other utilization and production facilities licensed pursuant to the Atomic Energy Act of 1954, as amended. Before a company can build a nuclear power plant at a particular site, it must obtain a construction permit from the NRC. Subsequently, the company must obtain an operating license from the NRC before it can operate the plant. The decision by NRC as to whether to approve a company's application for a construction permit or an operating license is based largely on the staff's detailed review of the information provided by the company as part of its application. This review responsibility also encompasses applications for approval of design certifications. Information provided by the applicant as part of the application is crucial to the licensing process as it provides NRC with the information it needs to make a decision with regard to the proposed plant's impact on the health and safety of the public. Information required by the NRC to be included in each application for a construction permit or an operating license is addressed in the Supporting Statements specific to the 10 CFR Part 50 Sections (see Enclosure 2).

2. Agency Use of Information

The NRC conducts a detailed review of all applications for licenses to construct and operate utilization and production facilities, in addition to applications for approval of design certifications. The purpose of the detailed review is to ensure that the proposed facilities can be built and operated safely at the proposed locations, and that all structures, systems, and components important to safety will be designed to withstand the effects of postulated accident conditions, without undue risk to the health and safety of the public. Applicants are required by the Act to provide such technical information and data that the NRC may determine necessary to ensure the public health and safety.

Part 50 of Title 10, Code of Federal Regulations, affects various types of facilities at various states in the licensing process. The requested information is reviewed and acted upon consistent with the governing NRC regulation or the Act, whichever is appropriate. For example, when a submittal can be completed without adjudication, the

collected information can usually be acted upon within 1 to 6 months. However, submittals which result in litigation may not be completed for 2 years or more.

Justifications for the number of copies required for various submittals, recordkeeping, and record retention requirements are not generally included in the separate Supporting Statements. Rather, generic justifications are provided in this transmittal statement.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to the use of information technology for reducing the burden associated with the information collection requirements specified in 10 CFR 50. Moreover, NRC encourages the use of such technology.

4. Effort to Identify Duplication

The information required by 10 CFR 50 does not duplicate other information collections required by other government agencies.

5. Effort to Use Similar Information

This information is available only from the licensees and applicants of utilization and production facilities.

6. Burden on Small Business

Certain provisions of 10 CFR 50 affect 48 critical facilities and research and test reactors operated by colleges and universities. However, most of the provisions affect only nuclear power plant licensees and applicants.

7. Consequences of Less Frequent Collection

This item is addressed in each Supporting Statement enclosed as Parts 1 through 26 (Enclosure 2).

8. Variation from OMB Guidelines

The subject information collection provisions generally do not vary from OMB Guidelines. However, see each part (Enclosure 2) for additional information.

9. Consultations Outside The NRC

Requirements of 10 CFR 50 are usually the subject of rulemaking proceedings, during which NRC received public comments. These comments were considered during the promulgation of all applicable final rules.

10. Confidentiality of Information

Information that is identified as proprietary or confidential, which is defined as information that, if disclosed, could do substantial harm with respect to (1) an organization's competitive position; (2) private and personal information; or (3) physical protection of safeguards information would be withheld from public disclosure pursuant to the provisions of 10 CFR 2.790 and 10 CFR 9.17.

11. Justification for Sensitive Questions

The provisions of 10 CFR 50 regulations generally do not require sensitive information. However, private information (e.g., telephone numbers) provided in Emergency Plans are protected in accordance with the provisions of 10 CFR 2.790 and CFR 9.17.

12.*

13.*

14. Reasons for Change in Burden

The estimated burden increase (from 3.7 M to 5.0 M hours) is primarily because during the 1986 review only 105 nuclear power plants were licensed; 111 plants are currently operating. There are three effective regulations involving additional burden that were not in place during the 1986 review: Licensee Notification, Bankruptcy Notification Requirements, and Decommissioning Requirements. Moreover, the increased number and type of generic issues currently being resolved involve more extensive industry reporting burden.

In addition, the estimated increase is also realized because of anticipated increase in reporting requirements related to Sections 50.36, 50.54(f), and 50.59(b); transfer of licenses; applications for termination of licenses; antitrust provisions; and the burden that was inadvertently omitted in 1986 on reactor startups.

15. Publication for Statistical Use

The information collected under 10 CFR 50 regulations is not published for statistical purposes.

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*Items 12 and 13 are covered in the section-specific statements (see Enclosure 2, Parts 1 through 26).

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Enclosures:
1. Table - Summary of Supporting Statements
2. Supporting Statements (Parts 1-26)
3. 10 CFR 50
4. Reference Publications

SUMMARY OF SUPPORTING STATEMENTS

10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	Estimated annual cost to industry	Estimated annual cost to federal government
1	50.30, 50.33 50.34, appl. for CP/OL	0	0	0	0	0	0	$147,200,000
	50.33a and Append. L, Anti-trust Review	100	2	20	180	200	$18,400	$61,364
	50.54(bb), Mgt. of Irrad. Fuel	0	0	0	0	0	0	0
	50.55b, Construction Completion	200	2	40	340	400	$36,800	$18,400
	Subtotals:	300	4	60	520	600	$55K	$147M

SUMMARY OF SUPPORTING STATEMENTS

10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	Estimated annual cost to industry	Estimated annual cost to federal government
1	50.59(c), 50.90, and 50.91(a) and (b), License Amend. Appl.	10	1,100	1,100	9,900	11,000	$1,012,000	$1,720,400
	50.65, Proposed Maintenance Rule	(Burden will be imposed when the rule becomes final)						
	50.74, Licensee Notification	1	400	40	360	400	$36,800	$36,600
	50.80(b), Transfer of Licenses	421	7	295	2,690	2,945	$270,940	$462,300
2	50.36, Tech Specs	772	718	234,720	319,770	554,490	$51,013,080	$3,910,736
3	Appendices, A&B 50.55(f)-QA	10,841	119	528,916	761,024	1,290,040	$118,683,680	$7,714,000
3a	50.55a, ASME Codes	92	119	10,948	0	10,948	$1,007,216	0
	Subtotals:	12K	2.5K	776K	1.0M	1.9M	$172M	$14M

SUMMARY OF SUPPORTING STATEMENTS
10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	Estimated annual cost to industry	Estimated annual cost to federal government
4	50.71, Bulletins and Generic Letters	10,844	119	129,520	1,165,676	1,295,196	$119,158,032	$2,553,920
5	50.48, Appendix R, Fire Protection	144	111	1,598	14,486	15,984	$1,470,528	$51,060
6	50.54(p), Security	200	290	5,800	52,200	58,000	$5,336,000	$2,612,800
7	50.47; 50.54(q, r, and t) Appendix E, Emergency Planning*	6,987	122	85,244	767,196	852,440	$78,424,480	$579,968
8	50.71(e) Updated FSAR	1,000	111	11,100	99,900	111,000	$10,212,000	$57,060
9	50.54(f) Oath or Affirm	2,084	172	35,850	322,650	358,500	$32,982,000	$754,400
		(5 plant specific; 48 res/test; 119 power)						
10	50.72 and 50.54(z), Notification of Events	10	111	111	999	1,110	$102,120	$9,814,800
10a	50.72(a), ERDS (Proposed Rule; burden will be included when rule is final)							
	50.73, (LERs)	(see OMB Clearance No. 3150-0104)						
	Subtotals:	21K	1.0K	269K	2.4M	2.7M	$247M	$16M

*See 50.72, Parts 10 and 10a for supporting statements pertaining to emergency notifications and responses.

SUMMARY OF SUPPORTING STATEMENTS

10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	Estimated annual cost to industry	Estimated annual cost to federal government
11	50.55(e) Design and Const. Deficiences	80	250 (reports)	2,000	18,000	20,000	$1,840,000	$184,000
12	50.59(b) Reports	2,000	159	254,400	63,600	318,000	$29,256,000	$1,170,240
13	Appendices G and H; 50.60, Fracture Toughness	113	83	940	2,460	9,400	$864,800	$356,960
14	Appendix J, Contain. Leakage	366	33 (reports)	1,208	10,870	12,078	$1,111,176	$3,036
15	50.35(b) Periodic Reports	0	0	0	0	0	0	0
15a	50.34(g), SRP Implementation	1,000	1	100	900	1,000	$92,000	$30,636
16	50.71(b) and Appendix C, Financial	1	119	12	107	119	$10,949	$10,948
17	50.54(w)(4) Property Damage Insurance	4	55	22	198	220	$20,240	$1,472
	Subtotals:	3.6K	700	259K	96K	361K	$33M	$1.8M

SUMMARY OF SUPPORTING STATEMENTS

10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	Estimated annual cost to industry	Estimated annual cost to federal government
18	50.44(c) Hydrogen Control	1,500	7	1,050	9,450	10,500	$966,000	$618,240
19	50.49, Environmental Qualification	234	119	9,520	18,380	27,900	$2,566,800	$73,232
20	50.62 ATWS	0	0	0	0	0	0	0
21	50.61 Pressurized Thermal Shock	110	73	805	7,245	8,050	$740,600	$39,253
22	50.64 Highly Enriched Uranium	402	7	281	2,532	2,813	$258,796	$191,520
23	Appendix K, 50.46, ECCS	39	113.6	445	4,008	4,453	$409,676	$288,595
	Subtotals:	2.3K	320	12K	42K	54K	$5.0M	$1.2M

SUMMARY OF SUPPORTING STATEMENTS

10 CFR 50

Part	Subject	Annual burden hours estimated per respondent	Number of responses annually	Estimated annual recordkeeping burden hours	Estimated annual reporting burden hours	Total estimated annual burden hours	estimated annual cost to industry	Estimated annual cost to federal government
24	50.63, Station Blackout	512	21	1,075	9,672	10,747	$988,724	$158,191
25	50.33(k), 50.75, and 50.82 Decommis-sioning Rule	1,147	8	5,904	3,272	9,176	$844,192	$126,408
26	50.54(cc), Bankruptcy Notifications	0	0	0	0	0	0	0
	Subtotals:	1.7K	29	7.0K	13.1K	20.0K	$1.8M	$285K
TOTALS:		4.5K	4.5.K	1.3M	3.6M	5.0M	$460M	$180M

Part 1

SUPPORTING STATEMENT

FOR

10 CFR 50.30, 50.33, 50.34,
50.54(bb), 50.55(b), 50.55(d), 50.59(c),
Proposed 50.65, 50.74, 50.80,
50.90, 50.91(a) and (b)

APPLICATION FOR CONSTRUCTION PERMIT OR OPERATING LICENSE
(AND OTHER MISCELLANEOUS SECTIONS OF 10 CFR PART 50)

DESCRIPTION OF INFORMATION COLLECTION

Applicants or licensees requesting approval to construct or operate utilization and production facilities are required by the Atomic Energy Act of 1954, as amended, (the Act), to provide information and data that the NRC may determine necessary to ensure the health and safety of the public.

Applications must contain information in three major categories to permit a complete evaluation by the NRC. These categories are general information, safety information which is submitted in two phases through a Preliminary Safety Analysis Report (PSAR) and a Final Safety Analysis Report (FSAR), and environmental information.

The section of the regulation that addresses each category of information for construction permit and operating license applications and NRC's detailed need within each category of information is outlined below. No applications for construction permits or operating licenses are anticipated during the next three years.

1. Construction Permit:

Section 50.30(a) provides for the filing of an application for a construction permit.

a. Contents of Applications:
General information (Sections 50.33, 50.33(f) and Appendix C). Here the applicant is identified and the financial qualifications are detailed.

Section 50.33(f) requires applicants to submit financial information that demonstrates reasonable assurances that required funds are available. Financial information is necessary because the NRC must make a decision as to whether the applicant's financial resources are adequate to permit construction of the plant in a safe manner and to permit implementation of safety-related programs described elsewhere in the

application. Appendix C of 10 CFR Part 50 outlines the information to be furnished by the applicant in the construction permit application to establish financial qualifications.

Information required for antitrust review must also be included in the construction permit application. The need for such information is addressed in Item 3 (Page 6).

b. Safety information (Sections 50.34, 50.34a, 50.34a(a), 50.34a(b), Appendix B, Appendix E). Safety information is provided by the applicant at the construction permit stage in the Preliminary Safety Analysis Report (PSAR). Section 50.34(a) outlines the minimum information that is necessary in the PSAR to permit the NRC to perform a safety evaluation. Included in the PSAR are the design criteria and preliminary design information for the proposed reactor and comprehensive data on the proposed site. The PSAR also discusses situations and the safety features which will be provided to prevent accidents or, if they should occur, to mitigate their effects on both the public and the facility's employees.

The principal features of the staff's safety review of the information provided in the PSAR by the applicant can be summarized as follows:

(1) A review is made of the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology. This review is necessary to determine whether these characteristics have been evaluated adequately and have been given appropriate consideration in the plant design and whether site characteristics are in accordance with NRC siting criteria.

(2) A review is performed of the facility design, and of programs for fabrication, construction and testing of plant structures, systems, and components important to safety for the purpose of determining whether they are in accord with the NRC regulations and other NRC requirements.

(3) A review is performed of the applicant's preliminary calculations of the response of the facility to a broad spectrum of hypothetical accidents for the purpose of determining whether site acceptability guidelines are satisfied.

(4) For the purpose of determining whether the applicant is technically qualified to operate the plant and whether he has established effective organizations and plans for continuing safe operation of the facility, a review is made of the applicant's plans for:

(i) plant operations including organizational structure,

(ii) technical qualifications of operating and technical support personnel,

(iii) planning for emergency actions to be taken in the event of an accident that might affect the general public (elements of preliminary planning that are required to be specified in the PSAR are set forth in 10 CFR 50.34(a) and Appendix E),

(iv) quality assurance (Appendix B) requires that the applicant provide in the PSAR, a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of safety-related structures, systems, and components.

(5) A review is made of the description of the preliminary design in systems to be provided by the applicant for control of radiological effluents from the plant. This review is necessary to evaluate the general adequacy of the systems proposed to control the release of radioactive wastes from the facility within the limits specified by the NRC regulations. Minimum information required by the NRC for this review is specified in Sections 50.34a(a) and 50.34a(b).

The NRC is currently reviewing safety analysis reports under 10 CFR 50.34 provisions for two standard design applications (General Electric's Advanced Boiling Water Reactor, and Combustion Engineering's System 80). The staff estimates that approximately 8,000 hours will be expended by the NRC annually to complete the review on each of these 2 applications. Thus, the annual estimated Federal cost is expected to be (16,000 hours x $92) $147,200,000.

c. Environmental Information. An Environmental Report, which provides a basis for the staff's evaluation of the environmental impact of the proposed plant, is specified as a requirement of the application for a construction permit in Section 50.30(f), but is justified in the OMB clearance for 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," (3150-0021).

d. 50.55(b), Construction Completion. If the proposed construction or modification of a facility is not completed by the latest completion date specified in the construction permit, the permit shall expire and all rights thereunder shall be forfeited. However, if good cause can be shown by the applicant the Commission may extend the completion date for a reasonable period of time. The Commission will recognize, among other things, developmental problems attributable to the experimental nature of the facility or fire, flood, explosion, strike, sabotage, domestic violence, enemy action, an act of the elements, and other acts beyond the control of the permit holder, as a basis for extending the completion date.

There are approximately 2 licensees who will be required to meet the regulations specified in 50.55(b) annually for the next 3 years. Preparing and filing the information that NRC needs in order to complete its review of requests for extension of construction permits will involve approximately 200 hours per licensee annually. This represents an annual industry cost of $36,800 (200 hours X 2 = 400 hours; 400 hours X $92 = $36,800).

Based on experience, NRC estimates that 100 staff hours will be involved for reviewing each of the 2 requests for construction permit extensions. This totals up to 200 annual person hours. Thus, annual Federal cost is expected to be $18,400 (200 hours X $92).

2. Operating License:

Pursuant to 10 CFR 50.55(d), at or about the time of completion of the construction or modification of the facility, the applicant must file any additional information needed to bring the original application or license up to date, and must file an application for an operating license or an amendment to an application for a license to construct and operate the facility for the issuance of an operating license, as appropriate, as specified in 50.30(d).

Section 50.30(d) provides for the filing of an application for an operating license. The information provided in this application is essentially an update of the information categories (i.e., general, safety, and environmental) previously submitted in the application for a construction permit.

a. General information (Section 50.33). Section 50.33(f) also requires applicants for operating licenses to submit financial information that demonstrates reasonable assurances that required funds are available. The applicant's financial qualifications must be detailed as they were for the construction permit application, but now the details must demonstrate that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license, plus the estimated costs of permanently shutting the facility down and maintaining it in a safe condition. The applicant shall submit estimates for total annual operating costs for each of the first 5 years of operation of the facility and estimates of the costs to permanently shut down the facility and maintain it in safe condition. The applicant shall also indicate the source(s) of funds to cover these costs. An application to renew or extend the term of an operating license must include the same financial information as is required in an application for an initial license.

b. Safety information (Sections 50.34(b), 50.34(c), 50.34(d), 50.34a(c), Appendix B, and Appendix E). Safety information is provided by the applicant at the operating license stage in the Final Safety Analysis Report (FSAR). Section 50.34(b) outlines the minimum information that should be provided in the FSAR to permit the NRC to perform a safety evaluation. This is essentially an update of information provided in the PSAR and allows the same editorial format. Among other things, the applicant must address the following items in the FSAR:

Pertinent details on the final design of the facility, including final containment design of the nuclear core and waste handling system; the applicant's latest plans for operation of the facility, as well as substantive procedures for coping with emergencies (Appendix E provides elements of emergency planning to be considered in the FSAR); the quality assurance program (Appendix B requires that information pertaining to managerial and administrative controls necessary to ensure safe operation of the plant be provided in the FSAR).

The final equipment design and procedures to be used by the applicant to control radiological effluents from the plant to permit the staff to determine whether such systems can control the release of radioactive wastes from the facility within the limits specified by NRC regulations. Information required by the NRC in the FSAR in this area of review is specified in Section 50.34a(c).

c. Physical Security Plan (Section 50.34(c))

This plan describes the physical program that will be provided in accordance with the requirements of Section 50.34(c) to assure that the plant will be sufficiently protected against acts of sabotage that could cause releases of radioactive materials in amounts sufficient to represent a hazard to the public health and safety. Also see Supporting Statement for 50.54(p), Part 6 to this 10 CFR 50 OMB package.

Safeguards Contingency Plan (Section 50.34(d))

The Safeguards Contingency Plan, as provided for in 10 CFR 50 will provide a structured, orderly, and timely response to safeguards contingencies and will be an important segment of NRC's contingency planning programs. Licensee safeguards contingency plans will result in organizing licensees' safeguard resources in such a way that, in the unlikely event of a safeguards contingency, the responding participants will be identified, their several responsibilities specified, and their responses coordinated.

d. Environmental Information. Justified in the Supporting Statement for 10 CFR Part 51, OMB Clearance No. 3150-0021.

Section 50.54(bb) requires that for operating nuclear power reactors, the licensee shall no later than 5 years before expiration of the reactor operating license, submit written notification to the Commission for its review and preliminary approval of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor upon expiration of the reactor operating license until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository. Final Commission review will be undertaken as part of any proceeding for continued licensing under Part 50 or Part 72. The licensee must demonstrate to NRC that the elected actions will be consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented on a timely basis. Where implementation of such actions require NRC authorizations, the licensee shall verify in the notification that submittals for such actions have been or will be made to NRC and shall identify them. A copy of the notification shall be retained by the licensee as a record until expiration of the reactor operating license. The licensee shall notify the NRC of any significant changes in the proposed waste management program as described in the initial notification.

Negligible burden is anticipated for this regulation because no reactor licensee is expected to be required to meet this provision during the duration of this 3-year clearance.

3. 50.33a and Appendix L, Information Requested by the Attorney General for Antitrust Review

Under the Act as well as other laws to protect trade and commerce against unlawful restraints and monopolies, the NRC is required to report promptly to the Attorney General any information it may have with respect to atomic energy which appears to violate or to tend toward violation of antitrust laws or to restrict competition in private enterprise. Further, upon request of the Attorney General, the NRC must furnish or cause to be furnished such information as the Attorney General determines to be appropriate for his advice on antitrust aspects of license applications for a utilization or production facility under Section 103 of the Act. The Attorney General's request is the basis for the NRC's antitrust reporting requirements.

The NRC staff estimates that two facilities will be required to meet the provisions of 10 CFR 50.33a and Appendix L while this clearance is in place. It is estimated that the effort expended by industry will involve approximately 300 hours for each facility. Annual industry burden and cost could, therefore, be 200 hours, and $18,400 (2 plants x 300 hours = 600 hours; 600 hours ÷ 3 = 200 hours annually x $92 = $18,400).

The Federal (NRC and Justice Department) effort is expected to involve approximately 1,000 hours per review, or a total of 2,000 hours. Annually, the Federal burden is, therefore, expected to involve approximately 667 hours, at an annual cost of approximately $61,364 (667 hours x $92).

4. 50.34(f) TMI Requirements

Requires that applications for operating licenses contain the Three Mile Island related requirements relative to the way the requirements will be implemented or satisfied prior to issuance of an operating license. These requirements include operational safety features, siting and design, and emergency preparedness, and are intended to provide substantial, additional protection in the operation of nuclear facilities based on experience from the accident at Three Mile Island and the various studies and investigations of the accident. Estimated burden for this requirement is zero because the NRC does not anticipate the submittal of applications for operating licenses during the duration of this clearance.

5. 50.59(c), 50.90, 50.91(a) and (b), Application for Amendment of License

Section 50.59(c) requires the holder of a license authorizing operation of a production or utilization facility who desires a change in technical specifications, or who desires a change in the facility or procedures described in the safety analysis report, or who desires to conduct tests or experiments which involve an unreviewed safety question to submit an application for amendment of the license. Section 50.90 requires the application for amendment of license or construction permit to be filed with the Commission, fully describing the changes and following as far as applicable the form prescribed for original applications.

The requirement for the amendment of the license application is needed to enable the staff to evaluate any changes made at the facility or any new

information concerning the facility that may potentially affect the safety of the facility and consequently the health and safety of the public. See the self-contained Supporting Statement prepared for 50.91(a) and (b), notification and State consultation, for the burden associated with this regulation (page 14).

6. Proposed 50.65, Maintenance Programs for Nuclear Power Plants

The NRC is proposing to amend its regulations to require commercial nuclear power plant licensees to strengthen their maintenance activities in order to reduce the likelihood of failures and events caused by the lack of effective maintenance. The NRC believes safety can be enhanced by defining an adequate maintenance program to ensure the effectiveness of such programs throughout the nuclear industry. The proposed rule would require plant maintenance programs to include specific activities, including the monitoring of the effectiveness of plant maintenance programs. The annual estimated burden on industry will be included in this Part 50 clearance when the rule is final.

7. 50.74, Licensee Notification to NRC

Section 50.74 requires licensees of nuclear power facilities to notify the NRC within 30 days of a change in status of a licensed reactor operator. It is estimated that there will be up to 400 respondents a year, that will involve 1 hour each of industry and NRC staff effort. Thus, the estimated burden for industry and the Federal government is expected to be $36,800 ($92 X 400).

8. 50.80(b), Application for Transfer of Licenses

NRC regulations in 10 CFR Part 50 establish requirements for the licensing of production and utilization facilities. The regulations were issued pursuant to the Atomic Energy Act of 1954, as amended, and Title II of the Energy Reorganization Act of 1974. Section 50.80, "Transfer of Licenses," specifies in paragraph 50.80(b) that an application for a transfer of a license shall include as much of the information described in sections 50.33 and 50.34 with respect to the identity and technical and financial qualifications of the proposed transferee as would be required by those sections if the application were for an initial license. Section 50.80(b) also specifies that the Commission may require additional information, such as data with respect to proposed safeguards against hazards from radioactive materials, and the transferee's qualifications to protect against such hazards.

The requirements described above are needed to assure the transferee's financial capability to run the facility safely and to ensure the transferee's technical capability to properly and safely operate the facility in a way that protects the health and safety of the public.

The NRC estimates that there will be approximately 5 applications annually that will be generally straightforward and will only involve effort pertaining to the transfer of minimal ownership percentages. These applications normally involve approximately 173 hours each of effort by industry and the NRC.

The NRC understands that approximately 5 licensees (Rancho Seco, Shoreham, Grand Gulf, Arkansas Units 1 and 2) plan to submit applications for transfer of license. We anticipate that approximately 2 will be submitted annually.

The review of these applications are expected to be extensive. Therefore, we believe review effort by the Federal government will encompass approximately 2,080 hours; licensee preparation of the applications is expected to involve approximately 1,040 hours.

Summary of Annual Burden and Cost, Section 50.80(b)

Federal government:

5 applications (ownership changes) x 173 hours = 865 hours

2 applications (new operating company) x 2,080 hours = 4,160 hours

865 hours + 4,160 hours = 5,025 hours; 5,025 hours x $92 = $462,300.

Industry:

5 applications (ownership changes) x 173 hours = 865 hours

2 applications (new operating company) x 1,040 = 2,080 hours

2,080 hours + 865 hours = 2,945 hours; 2,945 x $92 = $270,940.

A. JUSTIFICATION

1. Need for the Collection of Information

The U.S. Nuclear Regulatory Commission (NRC) is authorized by Congress to have responsibility and authority for the licensing and regulation of nuclear power plants, research and test facilities, fuel reprocessing plants and other utilization and production facilities licensed pursuant to the Act. To meet its responsibilities, the NRC conducts a detailed review of all applications for licenses to construct and operate such facilities. The purpose of the detailed review is to ensure that the proposed facilities can be built and operated safely at the proposed locations, and that all structures, systems and components important to safety will be designed to withstand the effects of postulated accident conditions, without undue risk to the health and safety of the public.

Before a company can build a nuclear power plant at a particular site, it must obtain a construction permit from the NRC. Subsequently, the company must obtain an operating license from the NRC before it can operate the plant. The decision by NRC as to whether to approve a company's application for a construction permit or an operating license is based largely on the staff's detailed review of the information provided by the company as part of its application. Information provided by the applicant as part of the application is crucial to the licensing process as it provides NRC with the information it needs to make a decision with regard to the proposed plant's impact on the public's health and safety. Information required by

the NRC to be included in each application for a construction permit or an operating license is addressed in the specific 10 CFR Part 50 sections for which this Supporting Statement, including those contained in Parts 1 through 26, is written.

"Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Regulatory Guide 1.70, Revision 3, indicates the information to be provided in the Safety Analysis Reports and represents a format for SARs that is acceptable to the NRC staff. Conformance with the Standard Format, however, is not required. Safety Analysis Reports with different formats will be acceptable to the staff if they provide an adequate basis for the findings requisite to the issuance of a license or permit. However, because it may be more difficult to locate needed information, the staff review time for such reports may be longer, and there is a greater likelihood that the staff may regard the report as incomplete.

2. Agency Use of Information

Upon receipt of an application, the NRC staff will perform a preliminary review to determine if the SAR provides a reasonably complete presentation of the information that is needed to form a basis for the findings required before issuance of a permit or license in accordance with 10 CFR Section 2.101. The Standard Format will be used by the staff as a guideline to identify the type of information needed unless there is good reason for not doing so. If the SAR does not provide a reasonably complete presentation of the necessary information, further review of the application will not be initiated until a reasonably complete presentation is provided. The information provided in the SAR should be up to date with respect to the state of technology for nuclear power plants and should take into account recent changes in the NRC regulations and guides and in industry codes and standards, results of recent developments in nulcear reactor safety, and experience in the construction and operation of nuclear power plants. The Standard Format should be used for both Preliminary Safety Analysis Reports and Final Safety Analysis Reports; however, any specific item that applies only to the FSAR will be indicated in the text by adding (FSAR) at the end of the guidance for that item. An entire section that is applicable only to the FSAR will be indicated by including (FSAR) following the heading.

The staff reviews in detail applications for construction permits and operating licenses to determine if the public health and safety will be fully protected. These reviews are conducted in some 50 different technical disciplines organized within the Office of Nuclear Reactor Regulation.

If any portion of an application is considered to be inadequate, the staff requests the applicant to make appropriate modifications or to provide needed additional information. In many cases, the staff review results in modifications to the facility's design or operating procedures. The result of the staff review is provided in a Safety Evaluation Report. This report represents a summary of the review and evaluation of the application by the staff relative to the anticipated effect of the proposed facility on the public health and safety. Safety Evaluation Reports are prepared for both the construction permit and operating license applications.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technology. Moreover, NRC encourages the use of technology.

4. Effort to Identify Duplication

The provisions of the 10 CFR Part 50 regulations are not required in other Federal regulations.

5. Effort to Use Similar Information

Licensees authorized to construct and to operate production or utilization facilities are the only source for the information.

6. Effort to Reduce Small Business Burden

Approximately 48 universities are affected by certain provisions of 10 CFR 50. The NRC is not considering any changes to these regulations, as mandated by the Atomic Energy Act of 1954, as amended, to reduce small business burden.

7. Consequences of Less Frequent Collection

The 10 CFR 50 regulations do not require that applications for construction permits or operating licenses be filed at a certain time.

8. Circumstances Which Justify Variation from OMB Guidelines

These information collections do not vary from OMB Guidelines.

9. Consultation Outside the NRC

Appendix L of 10 CFR Part 50 was developed in consultation with the Antitrust Division of the Department of Justice and has been amended twice at the request of the Department of Justice to refine the information needed for antitrust review.

Public comments are requested during the rulemaking phase of our final regulations. Since our last OMB review, we received comments on the Station Blackout rule (see Part 24) and the Decommissioning requirements (see Part 25).

10. Confidentiality of Information

Confidential or Proprietary Information is handled in accordance with the provisions of 10 CFR 2.790 and 10 CFR 9.17, "Agency Records Exempt from Public Disclosure."

11. Justification for Sensitive Questions

The provisions of 10 CFR 50 do not involve sensitive questions.

12. Estimated Cost to the Federal Government

The annual estimated cost to the Government is delineated at the end of the Summary Table enclosed with this statement.

13. Estimate of Industry Burden and Cost

See the Summary Table for applications for construction permit or operating license enclosed with this statement.

14. Reasons for Change in Burden

The burden reflected in this portion of the 10 CFR 50 OMB statement increased primarily because of the effectiveness of 10 CFR 50.74, reactor operators' notification to the NRC, and also because of the burden anticipated for (1) antitrust reviews encompassed within 10 CFR 50.33a and Appendix L, and (2) applications for transfer under 50.80(b).

15. Publication for Statistical Use

Information collected for review of applications for construction permit or operating licenses for utilization and production facilities is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the 10 CFR 50 information collections.

Enclosure:
Summary Table (Part 1)

SUMMARY TABLE
Application for Construction Permit or Operating License

(Part 1)

Subject	Annual Burden Hours Per Respondent	Number of Respondents Annually	Annual Recordkeeping Burden Hours	Annual Reporting Burden Hours	Total Annual Burden Hours	Annual Cost To Industry	Annual Cost to Federal Government
50.30, 50.33 50.34, 50.54(bb), et al.	0	0	0	0	0	0	$147,200,000*
50.55b, const. permit ext.	200	2	40	360	400	$36,800	$18,400
50.33a and Appendix L	100	2	20	180	200	$18,400	$61,364
50.34(f), TMI	0	0	0	0	0	0	0
50.59(c)** 50.90 and 50.91 (a) and (b), license amend. appl.	10	1,100	1,100	9,900	11,000	1,012,000	$1,720,400
50.74, licensee notification to NRC	1	400	40	360	400	$36,800	$36,800
50.80(b) transfer of license	421	7	295	2,690	2,945	$270,940	$462,300
Totals:	732	1,511	1,495	13,450	14,945	$1,374,940	$149,499,264

*This pertains to staff burden projected for completion of reviews for CESSAR-80 and GE's ABWR.
**See page 14 for supportive discussion.

SUPPORTING STATEMENT

"Notice for Public Comment: State Consultation," 10 CFR 50.91(a) and (b).

Justification

To expedite the public notification and State consultation procedures required by legislation, both NRC and the State need licensees' analyses and positions on significant hazards issues because licensees are in the best position to explain their amendment requests.

Description of Information Collection

Under §50.91(a)(1) and (b)(1) of Part CFR 50, a licensee requesting an amendment must provide to the NRC and the State in which its facility is located its amendment application and its analysis about the issue of significant hazards. In addition to requiring licensees' analyses of the license amendment requests, this section of the NRC's regulations also involves a reporting requirement concerning the issue of significant hazards consideration. The reporting requirement does not overlap or duplicate any other NRC or Federal information collection requirements. NRC needs licensees' analyses to quickly make and publish for public comment its "proposed determination" on significant hazards issues; and the States also need licensees' analyses in order to quickly consult with NRC.

Estimated Burden

The rule applies to 111 operating nuclear power plants and to two (2) testing facilities. Licensees of these reactors request about 1,100 license amendments per year. It is estimated that a licensee will spend approximately 10 hours per analysis under the standards in Section 50.92, "Issuance of Amendment." For 1,100 license amendment requests, the total burden on licensees would be 11,000 hours annually. Assuming an hourly rate of $92, an analysis request could cost a licensee about $920 (10 x $92). Thus, the total annual cost to industry for 1,100 amendment requests would be about $1,012,000 ($920 x 1,100 requests).

Estimated Cost to the Federal Government

NRC uses a licensee's analysis as a starting point for its significant hazards review. Including time spent in preparation of the Federal Register publication, NRC estimates that a total of 18,700 staff hours will be expended on 1,100 requests per year. This is derived from the staff's estimate that 20 percent of the .0415 staff-year per amendment request (17 hours) involves the significant hazards review and noticing in the Federal Register. Assuming an hourly rate of $92, for 1,100 amendment requests the cost to the government is estimated to be $1,720,400 (18,700 hours x $92).

SUPPORTING STATEMENT

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Part 2

SUPPORTING STATEMENT

FOR

10 CFR 50.36, 50.36a, 50.36b, AND APPENDIX I*

REPORTING AND RECORDKEEPING REQUIREMENTS CONTAINED
IN TECHNICAL SPECIFICATIONS CONTAINED IN LICENSES
TO OPERATE NUCLEAR POWER PLANTS

DESCRIPTION OF THE INFORMATION COLLECTION

Each licensee under 10 CFR 50 is required to perform reporting and recordkeeping requirements that NRC has approved as a part of the technical specifications (TS) submitted as a part of original applications for licenses. The reporting/recordkeeping requirements are set forth as "administrative controls" in Section 6 of the Appendix A Technical Specifications appended to each facility license. They are designed to ensure operation of the facility in a safe manner.

There are 111 operating nuclear power reactors and 48 operating research and test (non-power) reactors that are affected by the provisions of the captioned regulations.

A. JUSTIFICATION

1. Need for the Collection of Information

The typical reporting and recordkeeping burdens with justifications are explained below. NRC Regulatory Guide 1.16 (Revision 4), "Reporting of Operating Information--Appendix A Technical Specifications," provides the program being used by the NRC staff in order to standardize the reporting requirements section of Appendix A Technical Specifications of all operating licenses.

For licensees holding operating licenses without Appendix B environmental technical specifications or environmental protection plans, it may be necessary to include those reports identified in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," in the technical specifications under the unique reporting requirements section of the technical specifications.

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*Appendix I to 10 CFR 50 consists of numerical guides for design objectives and limiting conditions for plant operation to meet the criterion "as low as is reasonably achievable" for radioactive material in light-water cooled reactor effluents.

a. Radioactive Effluent Report

Section 50.36a of 10 CFR Part 50 specifies that to keep releases of radioactive materials to unrestricted areas as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor must include technical specifications. The NRC staff has developed "Radiological Effluent Technical Specifications for PWRs" (NUREG-0472) and "Radiological Effluent Technical Specifications for BWR's" (NUREG-0473). Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of the Procedural Details of RETS to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP)," permits relocation of the description of the radioactive effluent report content to the ODCM or the PCP. The contents of these three documents (as applicable) and the reporting requirements specified therein are being made part of the Appendix A Technical Specifications for new operating licenses. These same requirements are also being added to existing operating licenses as license amendments. (Appendix A Technical Specifications are approved by the NRC, are incorporated in the facility operating license, and are conditions of the license.)

Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation are to be submitted within 60 days after January 1 and July 1 of each year. This report includes a summary of the quantities of radioactive liquid and gaseous effluents released to the environment and solid waste shipped from the site. The staff has initiated a request for a rule change that will reduce the reporting frequency from semiannual to annual.

Special reports, or reports on exceeding design objectives, are required when certain conditions exist or parameters are exceeded, e.g., when the radiation dose for any calendar quarter is equal to or greater than one half the actual limit, or the annual dose exceeds twice the annual limit; when the liquid, gaseous or solid rad-waste treatment systems or the building ventilation systems are inoperable for more than 31 days.

b. Startup Report

Section 50.36, "Technical Specifications," of 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," requires that each applicant for a license authorizing operation of a nuclear power plant include in its application proposed technical specifications. These technical specifications, as approved by the NRC, are incorporated into the facility license and are conditions of the license. One of the reports normally required by the technical specifications is a startup report. This report is submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. The report addresses each test identified in the FSAR and should include a description of the test and the test conditions, the measured values of the operating condition or characteristics obtained during the test program, and a comparison of these values with design predictions and specifications.

The startup report provides the staff with evidence that the plant systems are functioning as designed and can be expected to perform as planned, in the safe operation of the plant.

The report is necessary to identify design deficiencies, and to obtain data on plant operation to verify (or provide a basis to modify) technical specification limits for operation. The data is also necessary for guidance in determining core reload requirements based on physics data obtained in testing reveal areas where additional performance verification testing is required or where further guidance is needed through additional regulatory guides or revision of existing guides.

c. Sealed Source Leakage Report

Section 50.36, 10 CFR Part 50, requires licensees to adhere to technical specifications for the construction and operation of production and utilization facilities. One specifically identified submission required of licensees by NRC under this authority is the Sealed Source Leakage Report, which includes technical specifications that establish requirements for testing the integrity of sealed sources transferred and for recording and reporting the tests results.

The reporting requirements on sealed sources licensed under 10 CFR Part 50 are included as a technical specification appended to the nuclear facility license. For some nuclear facility licenses, the reporting requirements for failed sealed sources require that a special report be submitted within 90 days following a test in which the results indicate removable contamination levels greater than 0.005mCi. Other nuclear facility licenses require reporting of such test results only as part of an annual report. Most reporting will be made annually, since any license that requires more frequent reporting can be amended, at the request of the licensee, to call for annual reports.

The information on any sealed source which exceeds the limitation on removable contamination should be reported annually for the licensed nuclear facility. If such information was not received, the quality assurance record for sealed sources used in operating a nuclear facility would be incomplete and failures would not be reported. Thus, the manufacturing process for maintaining the integrity of sealed sources under various operating conditions could be deficient, unknowingly.

The information obtained from nuclear facility licensees in Sealed Source Leakage Reports reflects a special type of use for sealed sources and provides further assurance that the manufacturing process can produce sealed sources with high integrity.

The staff is considering changes to the regulations that would delete this reporting requirement from the TS, and would move it to the category of documents that are maintained by the licensee. Therefore, the reporting requirement would no longer be mandatory.

d. Monthly Operating Report

Section 50.36, "Technical Specifications," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that each

applicant for a license authorizing operation of a commercial nuclear power plant include in its application proposed technical specifications. These technical specifications, as approved by the NRC, are incorporated into the facility license and are conditions of the license. One of the reports normally required by technical specifications is a report of operating statistics and shutdown experience. This report is submitted to the Commission by the licensees on a monthly basis. Information is submitted in the "Monthly Operating Report" regarding (1) Average Daily Unit Power Level, (2) Operating Data; (3) Unit Shutdowns and Power Reductions; and (4) Spent Fuel Storage Capacity, and is used as performance indicators.

Using the data from licensee's monthly reports, plus information received from NRC regional offices, the NRC prepares a monthly report, entitled "Operating Units Status Report." The report indicates, for each licensed unit, average daily power levels, operating status, unit shutdowns and power reductions, and summaries for all nuclear plant operations, including the capability to off-load spent fuel.

This monthly report is used by the NRC, the Department of Energy, and other Federal and State agencies. This report is necessary for Federal and State agencies to keep abreast of current plant operating data, including plant availability, which is of particular use during periods of reduced power output from other energy sources. Copies of the report are sent to the utilities to share with them the operating experience of other operators of nuclear power plants. The report is also available to the public.

The information obtained from the utilities is not otherwise available to the Federal Government on a current basis. Without this information Federal and State agencies could not keep abreast of current plant operations.

e. Non-Routine Environmental Reports

Environmental reviews of nuclear facilities often leave some questions only partially resolved. Data collection efforts authorized under 10 CFR Section 50.36b are intended to resolve these questions. Potentially significant environmental impacts (e.g., fish kills, excessive chemical releases, habitat disruption) need to be reported promptly so that appropriate action can be taken. To accomplish this result, Non-routine Environmental Reports are generally required by the technical specifications whenever an adverse effect may occur.

The non-routine report provides information which specifies and quantifies the data concerning the unusual events and provides the basis for recommending appropriate action. It provides the data in a timely fashion so that changes in operating procedures or design modifications can be implemented as soon as possible.

The NRC staff performs a detailed analysis of each event which warrants such study. The licensee's report and the NRC analysis are placed in the public document room and sometimes a press release is prepared. The staff analysis may recommend mitigative action.

There is no source for the required information other than licensees.

f. Annual Environmental Operating Report

Section 50.36 of 10 CFR Part 50 requires inclusion of technical specifications, based on analyses in the Safety Evaluation Report, in each license authorizing operation of a production or utilization facility.

Section 51.52 explicitly authorizes conditioning of a license to protect environmental values (e.g., commercial and sport fisheries, rare and endangered species, recreational land and water use). Non-radiological license conditions are generally incorporated in the license as Appendix B, Environmental Technical Specifications. The technical specifications discussed in Section 50.36 include requirements for an Annual Environmental Operating Report.

The purpose of nonradiological environmental monitoring is to confirm the environmental assessments presented in the Final Environmental Statement (FES) which described the impact of the proposed facility. The nonradiological programs are also designed to detect unanticipated adverse impacts (i.e., adverse impacts which exceed the predictions of the FES or impacts that were not predicted) soon enough to take appropriate action.

The operating procedures of a plant are sometimes conditioned to protect environmental values because of predictions in the FES that a potential for significant adverse impact exists. Monitoring programs are usually incorporated to assess the actual magnitude of predicted adverse impacts. If the impacts are different from those anticipated the licensee or staff can take action to change the technical specifications or plant design or operating procedures to more adequately account for the actual effects of facility operation.

g. Annual Radiological Environmental Operating Report

Section 50.36 of 10 CFR Part 50 provides that reactor operating licenses will include technical specifications which NRC finds appropriate. Each reactor license includes a technical specification requiring submission of annual radiological environmental operating reports.

The annual radiological environmental operating reports include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports also include the results of land use censuses required by the technical specifications and/or ODCM. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report provides an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports include summarized and tabulated results in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979,* of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report is submitted reporting and explaining the reasons for the missing results. The missing data are submitted as soon as possible in a supplementary report.

The report also includes the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by the Technical Specifications.

The report provides a record of environmental radiation around the plant. The report is reviewed by the NRC staff to determine whether radioactive material released routinely by nuclear power plants may have resulted in excessive environmental radiation. Without the report, the NRC staff could not provide adequate assurance that the public is being protected from such environmental radiation.

h. Annual Radiation Exposure Report

Section 50.36 requires that each applicant for a license authorizing operation of a nuclear power plant include in its application proposed technical specifications. These technical specifications, as approved by the NRC, are incorporated into the facility license and are conditions of the license.

The report, which is submitted annually, is a statistical summary of ranges of exposures. It provides data on sources of radiation exposure that provides key feedback to licensing and inspection programs related to radiation protection. The tabulation of occupational exposure data may be submitted along with any report of facility changes, tests or experiments, required pursuant to 10 CFR 50.59(b), or as a separate submittal at the option of the licensee.

The information on occupational personnel radiation exposure submitted by the licensees is necessary to enable the NRC staff to analyze procedures and hardware radiation exposure problems associated with operation, outage, or maintenance. The information provides a basis for evaluation of new plant designs or for modifications to present plant designs with respect to assuring that plants are designed for as low as reasonably achievable occupational radiation exposure.

Using data submitted by the licensees, the NRC also prepares an annual report entitled "Occupational Radiation Exposure at Commercial Nuclear Power Reactors" (NUREG-0713). Included in the report is a compilation

*This document pertains to the radioactive effluent reporting requirements discussed in paragraph a.

of in-plant occupational exposure data by work and job function. The information is required to establish trends among plants and within plants.

i. Recordkeeping Requirements

NRC Regulations in 10 CFR Part 50, Sections 50.36 and 50.36a establish requirements for recording results of reviews of events reported to the Commission and requirements for recordkeeping as part of administrative controls. The regulations were issued pursuant to the Atomic Energy Act of 1954, as amended, and Title II of the Energy Reorganization Act of 1974. These records are maintained primarily for the life of the plant. Certain records are only retained for 3 years (See Item 8).

Section 50.36(c)(1)(i)(A) requires recording of the results of reviews of events in nuclear reactors in which a safety limit has been exceeded. Section 50.36(c)(1)(i)(B) requires recording of the results of the reviews of events in fuel reprocessing plants in which a safety limit has been exceeded. Section 50.36(c)(1)(ii)(A) requires recording of the results of reviews of events in nuclear reactors in which an automatic safety system does not function as required. Section 50.36(c)(1)(ii)(B) requires recording of the results of reviews of events in fuel reprocessing plants in which an automatic alarm or protective device does not function as required. Section 50.36(c)(2) requires recording the results of reviews of events in nuclear reactors and fuel reprocessing plants in which a limiting condition for operation is not met. Each of the above records of review is required to include the cause of the condition and the basis for corrective action taken to preclude reoccurrence. Section 50.36(c)(5) requires administrative controls, including recordkeeping, in technical specifications of a production or utilization facility as necessary to assure operation of the facility in a safe manner. Details of recordkeeping are delineated in Section 6.10 of Standard Technical Specification, NUREG-0123 for General Electric boiling water reactors, NUREG-0212 for Combustion Engineering pressurized water reactors, NUREG-0103 for Babcock and Wilcox pressurized water reactors and NUREG-0452 for Westinghouse pressurized water reactors.

The records required by Section 50.36(c)(5) involve such matters as:

(1) Records and logs of facility operation covering time interval at each power level.

(2) Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

(3) All reportable events.

(4) Records of surveillance activities, inspections and calibrations required by the Technical Specifications.

(5) Records of changes made to Operating Procedures.

(6) Records of radioactive shipments.

(7) Records of sealed source and fission detector leak tests and results.

(8) Records of annual physical inventory of all sealed source material of record.

(9) Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

(10) Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

(11) Records of facility radiation and contamination surveys.

(12) Records of radiation exposure for all individuals entering radiation control areas.

(13) Records of gaseous and liquid radioactive material released to the environs.

(14) Records of transient of operational cycles for various facility components.

(15) Records of reactor tests and experiments.

(16) Records of training and qualification for current members of the plant staff.

(17) Records of in-service inspections performed pursuant to the Technical Specifications.

(18) Records of Quality Assurance activities required by the QA Manual.

(19) Records of reviews performed or changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR Part 50, Section 50.59.

(20) Records of meetings of safety review groups.

(21) Records of the service lives of all snubbers required by the Technical Specifications.

(22) Records of secondary water sampling and water quality.

(23) Records of analyses required by the Radiological Environmental Monitoring Program.

These records are used by the licensees, the NRC and other Federal, State and local government agencies for the review of a variety of activities in the facility, many of which affect safety. The records are also historical in nature and provide data on which future activities can be based. NRC Inspection and Enforcement personnel can spot

check the records required by 50.36 to determine, for example, if (1) plant modifications were performed satisfactorily, (2) the plant was operated within the technical specifications, (3) personnel training has been kept current, (4) plant effluents have been kept within allowable values, etc. Because of the multiple-use nature of many of the records, NRC has estimated only the incremental burden.

2. Agency Use of Information

NRC uses this information to determine whether releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, are as low as is reasonably achievable. The Agency also uses this information to ensure the protection of the non-radiological environment.

Moreover, safety limits, limiting safety system settings, and limiting control settings, limiting conditions for operation, surveillance requirements, and design features are monitored by the technical specifications for ensuring that the health and safety of the public are not affected adversely from the operation of nuclear power reactors.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through the use of information technologies, and NRC encourages its use.

4. Effort to Identify Duplication

Some duplication of Agency requirements has been identified. The TS are being revised to limit duplication, reduce burden, and clarify requirements.

5. Effort to Use Similar Information

There is no source for the required information other than licensees.

6. Effort to Reduce Small Business Burden

There are only 48 non-power reactors subject to the provisions of the technical specification regulations. The affected universities are encompassed within the Technical Specification Improvement Program currently ongoing at the NRC. The burden for non-power reactors cannot be further reduced without potentially affecting the health and safety of the public.

7. Consequences of Less Frequent Collection

The NRC would not be able to ensure that the health and safety of the public is not adversely affected by the operation of a reactor.

8. Circumstances Which Justify Variation from OMB Guidelines

The records involved with this information collection are maintained primarily for the life of the plant in order to ensure that the health and safety of the public will not be affected adversely by the operation of the plant.

9. Consultation Outside the NRC

Requests by the licensees for changes to the technical specifications are noticed in the Federal Register. Public comments or requests for hearing are considered by the NRC.

10. Confidentiality of Information

Proprietary or confidential information is handled in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

The subject regulations do not request sensitive information.

12. Estimate of Cost to the Federal Government

Estimated hours of staff effort involved for the review of each report is delineated below.

1. Radioactive Effluent Report

a. Exceeding Design Objectives - 3 reports are anticipated annually x 40 hours of staff review = 120 hours.

b. Semi-Annual Effluent Reports - 111 respondents will submit these reports twice a year; 111 respondents x 2 reports/licensee x 40 hours = 8,880 hours.

The total staff effort for these reports would be 9,000 hou

2. Startup Reports

This reporting requirement will affect approximately 46 licensees annually. Approximately 80 staff hours will be needed to review these reports. Thus, the annual estimated burden to the NRC is expected to be (46 reports x 80 hours) 3,680 hours.

The estimated burden decreased from 120 to 80 hours because based on the current staff's experience, and the status of various technical issues, it is believed that only 80 hours are needed to review each report.

3. Sealed Source Reports

Approximately 10 reports are expected to be filed annually. Approximately 8 hours will be needed to assess the reports. Therefore, approximately 80 hours of staff effort is expected.

4. Monthly Operating Report

Twelve (12) reports are submitted annually by 111 licensees. The staff assesses each of these reports in approximately 8 hours. The burden, therefore, would be 10,656 hours (111 x 12 = 1,332 reports x 8 hours = 10,656 hours).

5. Non-routine Environmental Report

One (1) report is submitted annually for 74 sites. The staff's efforts to assess these reports is usually about 40 hours each. Thus, the burden for the staff is expected to be 2,960 hours (74 x 40).

6. Annual Radiological Environmental Operating Report

This report will be submitted for 74 sites. It is estimated that approximately 173 hours will be needed to review this report. Therefore, the staff burden is estimated to be 12,802 hours (173 hours x 74 sites).

7. Annual Environmental Operating Report

This report contains information on non-radiological environmental effects (e.g., cooling tower blowdown) and is normally called for in the old Appendix B license requirements. Thus, the Federal burden associated with this report is expected to be negligible.

8. Annual Radiation Exposure Report

It is estimated that the staff will assess each report within 30 hours. One hundred eleven (111) licensees will respond annually. Thus, the burden is expected to be 3,330 hours (30 x 111).

Total Federal Burden and Cost

Based on the above, the estimated staff burden would be 42,508 hours at a cost of $3,910,736 ($92 x 42,508 hours).

13. Estimate of Industry Burden and Cost

a. Radioactive Effluent Reports

These include reports on (a) Exceeding Design Objective Doses, (b) Inoperable Radwaste Equipment, (c) Dose Contribution from Effluents, (d) Unplanned Radioactive Release, (e) Exceeding 10 CFR Part 20 Release Limits and (f) Exceeding Ci Content in Liquid or Gaseous Tanks or Ci Release Rate for Offgas System (BWR), which involve approximately 50 hours each for about 3 licensees (a total of about 150 hours); and a Semi-Annual Effluent Report which requires each of 111 licensees 140 hours per report for a total burden of 31,080 hours annually (111 x 140 = 15,540 x 2 reports = 31,080 hours).

Therefore, the total estimated burden for these reports is expected to be 31,230 (150 + 31,080 hours).

b. Startup Report

It is estimated that this reporting requirement will affect approximately 46 licensees annually with an average burden of 140 hours per licensee. The burden is, therefore, estimated to be 6,440 hours.

The estimated industry burden increased from 100 to 140 hours because it is believed that details pertaining to certain current safety-related issues involve more analytical burden.

c. Sealed Source Report

Since the licensee will be required to report only those sealed source test results which exceed the removable contamination limit, burden will be negligible, approximately 10 licensees will be affected. (160 hours, assuming 16 hrs/report).

d. Monthly Operating Report

One hundred eleven (111) licensees each submit 12 reports annually, each report imposing a burden for preparation of 50 hours.

111 x 12 x 50 hours total 66,600 burden hours.

e. Non-routine Environmental Report

An average of about one report is submitted for each site annually. Thus, the preparation burden (50 hours per report), assuming 74 sites (50 x 74) would be 3,700 hours.

f. Annual Radiological Environmental Operating Report

Licencees will submit this report for an estimated 74 sites in response to this requirement. The report causes a preparation burden of 1400 man-hours. This results in 103,600 burden hours (1400 hours x 74 sites).

g. Annual Environmental Operating Report

Licensees for 74 sites are required to submit this report. Each report could require approximately 1,400 hours to prepare, which would result in 103,600 hours of burden annually.

h. Annual Radiation Exposure Report

The estimated burden upon each power reactor licensee for the preparation of one report is 40 hours.

111 x 40 hours totals 4,440 staff-hours.

The total estimated reporting burden for all licensees is 319,770 hours.

ESTIMATE OF RESPONDENT RECORDKEEPING REQUIREMENTS

These recordkeeping requirements called for under 50.36(c) are subject as follows:

111 operating reactors
48 research/test reactors (non-power)

The burden annually for an operating power reactor is estimated to be approximately 2,080 hours. One hundred eleven (111) operating power plants x 2,080 hours totals 230,880 hours.

The burden annually for a research or test reactor is estimated to be approximately 80 hours. Forty-eight (48) research/test reactors (non-power) x 80 hours totals 3,840.

Total for recordkeeping burden of all licensees: 234,720 hours (230,880/power + 3,840/research).

TOTAL INDUSTRY BURDEN AND COST

Total annual burden for all reporting/recordkeeping requirements for technical specifications is expected to be 554,490 (319,770 reporting + 234,720 record-keeping) hours. The total annual cost to industry at $92 per hour would be $51,013,080.

14. Reasons for Change in Burden

The estimated annual burden increased from 372,185 to 450,890 hours because there are 111 operating power reactors; 105 were licensed to operate during the last OMB review. In addition, the previous statement inadvertently omitted startup reports for the refueling plants.

15. Publication for Statistical Use

The collected information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Attachment:

Table - Burden and Cost Summary

TABLE - ANNUAL BURDEN AND COST SUMMARY
10 CFR 50.36, TECHNICAL SPECIFICATIONS

Item	Estimated respondents	Estimated burden per respondent	Estimated reporting burden hours	Total industry burden hours	Estimated industry cost	Estimated Federal burden hours	Estimated Federal cost
a. Effluent Reports							
- Exceed. Design Doses, etc.	3	50	150	150	$13,800	120	$11,040
- Semi-Annual Effluent	111 (2 reports each)	140	31,080	31,080	$2,859,360	8,880	$816,960
b. Startup Report	46	140	6,440	6,440	$592,480	3,680	$338,560
c. Sealed Source	10	16	160	160	$14,720	80	$7,360
d. Monthly Oper.	111 (12 reports each)	50	66,600	66,600	$6,127,200	10,656	$980,352
e. Non-Routine Environ.	74	50	3,700	3,700	$340,400	2,960	$272,320
f. Annual Radiolog.	74	1,400	103,600	103,600	$9,899,200	12,802	$1,177,784
g. Annual Environ.	74	1,400	103,600	103,600	$9,899,200	0	0
h. Annual Rad. Exposure	111	40	4,440	4,440	$408,480	3,330	$306,360
i. Records called for in 50.36(c)	158	1,486	0	234,720	$21,594,240	0	0
TOTALS:	772	4,772	319,770	554,490	$51,013,080	42,508	$3,910,736

*111 power and 48 non-power reactors are required to maintain these records

Part 3

SUPPORTING STATEMENT
FOR
QUALITY ASSURANCE (QA) RECORDS

Section 10 CFR 50.55a,* 50.55(f), Appendix A (Criterion 1), and Appendix B

DESCRIPTION OF THE INFORMATION COLLECTION

QA records associated with the activities listed below are used by the licensee, the National Board of Boiler and Pressure Vessel Inspectors, insurance companies and the NRC in the review and confirmation of quality related activities. Most States and all nuclear insurers already require that the ASME B&PV Code (Section III) be used in the design, construction, testing and inspection of nuclear power reactors.

Appropriate records of the design, fabrication, erection and testing of structures, systems and components important to safety shall be maintained by the licensee throughout the life of the plant.

1. Management: QA manual, procedures, and instructions
2. Qualification and training of personnel
3. Design
4. Procurement, items identification/control, acceptance status
5. Manufacture, installation/testing
6. Handling, storage and shipping
7. Inspection, testing and qualifying, including inspection status
8. Calibration
9. Special processes
10. Operation
11. Maintenance
12. Modification and repair
13. Audits
14. Non-conformance, corrective actions

A. JUSTIFICATION

1. Need for the Collection of Information

Licensee burden hours will be spent on QA records development and maintenance, which pertain to Items 1 through 14 listed above.

Appendix B requires records for "safety-related" items that are usually found on a plant-specific Q-list. These record requirements were the basis for the burden hours previously reported to allow for the additional QA records required by Appendix A (but not prescribed by the NRC) in connection with items which are not "safety-related".

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*See Part 3a, separate Supporting Statement for information collection requirements specified in 10 CFR 50.55a.

Regulatory Guide 1.28 (Rev. 3) "Quality Assurance Program Requirements (Design and Construction)" describes an acceptable method for complying with QA records requirements in accordance with 10 CFR Part 50. Except for a few regulatory positions in the Regulatory Guide, it endorses the common industry standard ANSI/ASME NQA-1-1983, "Quality Assurance Program Requirements for Nuclear Facilities." Maintenance of records as specified above is necessary so that evidence can be furnished to show that activities affecting quality have been accomplished in accordance with NRC regulations.

The type of records identified specifically in Criterion XVII of Appendix B to 10 CFR Part 50 are of particular importance to provide adequate evidence that licensee activities affecting quality have been accomplished in accordance with NRC regulatory requirements. Records pertaining to items which are not "safety-related" are not detailed in any specific NRC requirements document but are, nevertheless, expected to be available for inspection and audit by the NRC in accordance with Criterion 1 of Appendix A to 10 CFR Part 50.

Reporting of changes to the QA program pursuant to 10 CFR Part 50.55(f) has been a requirement since March 1983. The licensee's QA Program Plan, after acceptance by the NRC, is a license condition. Any changes to this plan must be reported to the NRC like other license conditions of a similar nature. It is estimated that each licensee/applicant will initiate one such change per year. Such changes are included in the total license amendment requests reflected in the Supporting Statement that covers 50.91(a) and (b) at page 14.

2. Agency Use of Information

Records required to be maintained for a specific activity are specified in the license application, license condition or permit, or NRC-approved documents. These records, some of which will be kept for the life of the facility, are available for inspection by the NRC, and are reviewed and examined to acertain whether the activities affecting quality have been accomplished in accordance with NRC requirements. Also, in case of malfunction or failure of an item affecting safety, availability of plant records is necessary to aid in the determination of the cause of the failure. In addition, records maintenance is necessary for other important specific functions such as providing baseline data for inservice inspection and providing data for trend analysis.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through the uses of information technology.

4. Effort to Identify Duplication

These records are not a duplication of any other records maintained by the licensee or Federal Government.

5. Effort to Use Similar Information

This information is only available from the licensee.

6. Effort to Reduce Small Business Burden

The subject provisions do not affect small businesses.

7. Consequences of Less Frequent Collection

QA records are collected as they are generated during design, construction, and operation of the plants. Less frequent collection is not an alternative.

8. Variation from OMB Guidelines

The records must be retained throughout the life of the plant in order to ensure the health and safety of the public.

9. Consultation Outside the NRC

None

10. Confidentiality of Information

This information is usually not confidential. If it were, the information would be handled in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

These regulations do not require sensitive information.

12. Estimated Cost to the Federal Government

QA records are generated and maintained by licensees. The incremental cost of NRC audits and inspection of QA records is a small part of the total NRC inspection program, consisting of the resident inspectors, the regional inspections, and the special inspections. It is estimated that 6.5 percent of the licensee's burden hours are necessary for NRC audit and inspection (.065 x 1,290,040 = 83,850 staff hours). This estimate is based on FTE loading of NRC inspection activities and estimates of inspectors' record review activities gained from interviews and from personal experience of the NRC staff representative.

Therefore, the estimated Federal cost is expected to be $7,714,000 ($92 x 83,850).

13. Estimate of Industry Burden and Cost

a. Estimated Reporting Burden:

Each of 8 plants under construction generates a licensee burden of 20,000 burden hours (8 x 20,000) = 160,000 hrs/yr

Each of 111 operating reactors generates a licensee burden of 10,000 burden hours per year (111 x 10,000) = 1,110,000 hrs/yr

Reporting changes to the QA Program for each of the four large test reactors causes the licensee to expend 250 hours per year; 4 x 250 = 1,000 hrs/yr

Reporting changes to the QA Programs, 119 licensees x 160 burden hours (1 report from each licensee; 160 hours of burden for each) 19,040 hrs/yr

Total Burden Hours: 1,290,040 hrs/yr

Cost is based on $92.00 per hour for licensee; therefore, cost to industry = $118,683,680

b. Estimated Recordkeeping Burden

A comprehensive system of planned and periodic audits must be carried out by licensees to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits are performed in accordance with quality assurance program procedures. Based on NRC's experience and in light of the magnitude of records for the audits and the overall program during construction, it is estimated that 41 percent of the total industry reporting burden (1,290,040 hours) encompasses hours expended annually for recordkeeping requirements. Recordkeeping requirements are, therefore, estimated to involve 528,916 hours annually.

14. Reasons for Change in Burden

The burden hours have been reduced from 1.5 million hours to 1.3 million hours because the number of plants under construction has been reduced from 22 to 8 since the last OMB review.

15. Publication for Statistical Use

The information collected under the QA provisions of the NRC regulations is not published for statistical use.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the collection of the subject information.

Supporting Statement for Information Collection Requirements in

10 CFR 50.55a

(ASME CODE)

DESCRIPTION OF INFORMATION COLLECTION

NRC Regulations in 10 CFR 50.55a incorporate by reference Division 1 rules of Section III, "Rules for Construction of Nuclear Power Plant Components," and Division 1 rules of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). These sections of the ASME Code set forth the requirements to which nuclear power plant components are designed, constructed, tested and inspected. Inherent in these requirements are certain recordkeeping functions.

In general, the information collection requirements incurred by §50.55a through endorsement of the ASME Code apply to the owners of the 8 nuclear power plants under construction and to the owners of the 111 nuclear power plants with operating licenses. The actual number of plants that would implement the edition and addenda addressed by the proposed revision, and thereby be affected by their information collection requirements, is dependent on a variety of factors. These factors include whether the application is for Section III or Section XI, the class and type of components involved, the date of the construction permit application, the schedule of the inservice inspection program, and whether the plant voluntarily elects to implement updated editions and addenda of the ASME Code.

NRC applicants and licensees have been complying with the information collection requirements of the ASME Code since 1971. No problems with these information collection requirements have been identified to the NRC by the applicants or licensees.

The retention period for information is in accordance with a schedule provided in Table NCA-4134.17-1 and paragraph IWA-6300 of the ASME Code. The record retention periods are:

Information	Retention Period
Procedure for attachment of nameplates.	3 years after superseded
Overpressure protection report.	Lifetime[1]
Welding procedure specification.	Lifetime
NF-1/N-5 Data Report Forms.	Lifetime
Design Report for storage tanks.	Lifetime
Test results for vacuum relief valves.	Lifetime
Proced. for control of welding material.	3 years after superseded
Preservice Inspection Plan.	Lifetime
Ultrasonic examination records.	Lifetime

[1]Service lifetime of the component or system.

Incorporation of the Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Section III, Division 1, of the ASME Code adds the following recordkeeping requirements. These records are not collected but are retained by the licensee to be made available to the NRC in the event of a routine NRC audit.

Section III

- Winter 1984 Addenda

  NCA-8213(c); Attachment of Nameplates - Nameplates attached with pressure sensitive acrylic adhesive system must meet requirements of mandatory Appendix XXI, "Adhesive Attachment of Nameplates," which stated that Certificate Holders QA manuals contain written procedures for the application of nameplates.

  NB/NC/ND-7441; Installation of Pressures Relief Devices - Technical justification for the adequacy of the installation of such devices must be provided in the Over-pressure Protection Report.

- Summer 1985 Addenda

  NB-4622.11; Temper Bead Weld Repair to Dissimilar Metal Welds or Buttering - Requires preparation of welding procedure specification.

- Winter 1985 Addenda

  NCA-8440; Data Reports, Components Supports - New provisions require the use of NF-1 and N-5 Data Report Forms.

  NC-3812/NC-3912; Design Report - Certificate Holder must provide Design Report as part of his responsibility for achieving structural integrity of atmospheric storage tanks (NC-3812) and 0 - 15 psi storage tanks (NC-3912).

- Winter 1984 Addenda

  No additional information collections

  NC/ND-7750; Capacity Certification of Vacuum Relief Valves - Requirement that drawings and test results are to be submitted to the ASME designee for review and acceptance.

- 1986 Edition[2]

  The Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, and 1983 Edition have been incorporated by reference into 10 CFR §50.55a by a previous amendment. Information collection requirements for the Winter 1984 Addenda, Summer 1985 Addenda, and Winter 1985 Addenda are discussed above.

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[2]The 1986 Edition of Section III, Division 1, is equivalent to the 1983 Edition, as modified by the Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, and the Winter 1985 Addenda.

Incorporation of the Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Section XI, Division 1, of the ASME Code adds the following recordkeeping requirements.

Section XI

- Winter 1983 Addenda

  IWA-2420; Inspection Plans and Schedules - Requires preparation of preservice and inservice inspection plans and schedules.

  IWA-4210; Storage and Handling of Welding Material - Requires written procedure for storage, handling, and control of welding material.

  IWA-6210(a)/(c); Owner's Responsibility - Requires preparation of preservice inspection plans and schedules, records, and summary report for Class 1 and 2 pressure retaining components and their supports.

  IWA-6340(b)/(c); Inservice Inspection Records and Reports - Requires maintenance of preservice inspection plans, schedules, and reports.

  Appendix VI (Mandatory); Ultrasonic Examination of Bolts and Studs - requires records of the personnel performance qualification tests, written procedures for ultrasonic examination, documentation of the procedure qualification, calibration records, and examination records.

- Summer 1984 Addenda

  No additional information collections

- Summer 1985 Addenda

  No additional information collections

- Winter 1985 Addenda

  IWA-2232(f); Ultrasonic Examination - Indications that can be identified as originating from surface configurations or variations in metallurgical structure of materials may be classified as geometric indications. The maximum amplitude, location, and extent of a geometric indication shall be recorded.

The NRC has issued a proposed rulemaking that will update existing references to specific sections of the ASME Boiler and Pressure Vessel Code that set forth the requirements by which nuclear power plant components are constructed, inspected, and tested. Specifically, the proposed rulemaking will incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Division 1 rules of Section III, "Rules for Construction of Nuclear Power Plant Components," and Division 1 rules of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The estimated burden associated with this rulemaking will be reflected in this statement when the provisions are incorporated in 10 CFR 50.55a as final regulations.

- 1986 Edition[3]

  The summer 1983 Addenda and 1983 Edition have been incorporated by reference into 10 CFR §50.55a by a previous amendment. Information collection requirements for the Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, and Winter 1985 Addenda are discussed above.

A. JUSTIFICATION

1. Need for the Collection of Information

The ASME Code provides listings of information required and specific forms to assist in documenting required information. The information is generally not collected, but is retained by the licensee to be made available to the NRC in the event of an NRC inspection or audit. ASME Code requirements are incorporated in 10 CFR 50 to avoid the need for writing equivalent NRC requirements.

2. Agency Use of Information

These records are used by the licensees, National Board inspectors, insurance companies, and the NRC in the review of a variety of activities that affect safety. The records are generally historical in nature and provide data on which future activities can be based. NRC inspection and enforcement personnel can spot check the records required by the ASME Code to determine, for example, if proper inservice examination methods were utilized.

3. Reduction of Burden Through Information Technology

The information being collected represents the documentation for the various plant-specific construction, inservice inspection, and inservice testing programs. The NRC has no objection to the use of new information technologies and encourages its use.

4. Effort to Identify Duplication

ASME Code requirements are incorporated by reference into the NRC regulations to avoid the need for writing equivalent NRC requirements. The provisions of this regulation do not encompass other generic regulatory requirements.

5. Effort to Use Similar Information

The NRC is using the information reporting requirements specified in the ASME Code in lieu of developing its own equivalent requirements.

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[3]The 1986 Edition of Section XI, Division 1, is equivalent to the 1983 Edition, as modified by the Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, and the Winter 1985 Addenda.

6. Effort to Reduce Small Business Burden

The provisions of 10 CFR 50.55a only involve nuclear power plants and, therefore, do not affect small companies.

7. Consequences of Less Frequent Collection

The information is generally not collected, but is retained by the licensee to be made available to the NRC in the event of an NRC audit.

8. Circumstances Which Justify Variation from OMB Guidelines

The record retention period is frequently for the service lifetime of the applicable component. Such lifetime retention of the records is necessary to ensure adequate historical information of the design and examination of components and systems to provide a basis for evaluating degradation of these components and systems at any time during their service lifetime.

9. Consultations Outside the NRC

Estimates of burden were obtained, at the time the original supporting statement was prepared, from inservice inspection staff and consultants to utilities, and national lab personnel engaged in inservice inspection activities. However, there were no consultations during this update period.

10. Confidentiality of Information

NRC provides no pledge of confidentiality for this collection of information. However, a confidential or proprietary submittal would be handled in accordance with 10 CFR Section 2.790.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of Annual Cost to the Federal Government

NRC inspection personnel who audit plant construction, inservice inspection, and inservice testing records as part of the routine NRC inspection process would include in their audit verification that the above records are being properly prepared and maintained. The time associated with NRC inspectors verifying these records would be extremely small when the activity is performed as part of the normal NRC inspection process.

13. Estimate of Annual Industry Burden

a. Estimated Hours

The information collection requirements inherent in incorporating by reference the latest edition and addenda of Section III, Division 1, and Section XI, Division 1, of the ASME Code are identified above in

"Description of Information Collection". These requirements may be categorized in terms of Section III requirements that document component design, and Section XI requirements that address ultrasonic examinations and preservice inspection programs.

The additional Section III requirements incur a one-time burden on plants under construction. Section 50.55a specifies that the Code Edition, Addenda, and optional Code Cases to be applied to reactor coolant pressure boundary, and Quality Group B and Quality Group C components must be determined by the provisions of paragraph NCA-1140 of Subsection NCA of Section III of the ASME Code. NCA-1140 specifies that the owner (or his designee) shall establish the ASME Code edition and addenda to be included in the Design Specifications, but that in no case shall the Code edition and addenda dates established in the Design Specifications be earlier than three years prior to the date that the nuclear power plant construction permit application is docketed. NCA-1140 further states that later ASME Code editions and addenda may be used by mutual consent of the Owner (or his designee) and Certificate Holder. The earliest Section III addenda addressed in the subject regulation is the Winter 1984 Addenda. Since the last plants docketed in October 1974 (Palo Verde Units 1, 2, 3), there is no plant under construction for which implementation of the Section III edition and addenda specified in the rule is a requirement. Plants may implement these improved rules on a voluntary basis, but unless they make that choice, there is no additional paperwork burden associated with incorporating the proposed Section III edition and addenda.

The additional Section XI requirements incur an information collection burden associated with the documentation of procedures for the storage, handling, and control of welding material; the documentation of preservice inspection plans and schedules; and the implementation of the ultrasonic examination of bolts and studs.

Nuclear power plants are required to update their inservice inspection programs by incorporating into their initial 120-month inspection interval requirements of the latest edition and addenda of Section IX, Division 1, that have been incorporated by reference into §50.55a as of 12 months prior to the date of issuance of operating license; and by incorporating into successive 120-month inspection intervals requirements of the latest edition and addenda of Section XI that have been incorporated by reference as of 12 months prior to the start of a 120-month inspection interval. On this basis, many plants will at one time be required to implement the Section XI, Division 1, edition and addenda specified in the regulation. The number of plants that will be implementing the specified edition and addenda will grow gradually as each plant updates it inservice inspection program at the 10-year interval. Therefore, conservatively, the total number of plants that may ultimately be required to implement the specified edition and addenda is 119 (i.e., 111 operating plants and 8 plants under construction).

Paragraph IWA-2420 was revised in the Winter 1983 Addenda to require the preparation of preservice and inservice inspection plans, to specify the content of those plans, and to require an implementation schedule for the performance of examination and tests. Inservice inspection plans and schedules, including those for preservice inspection, are routinely submitted to the NRC as part of the regulatory process. The types of items specified for inclusion in the plan (e.g., applicable Code edition and addenda, identification of components subject to examination and test, and Code requirements for each specified component) are typically included in the plans submitted to the NRC. Therefore, there is no additional recordkeeping burden associated with the revision of IWA-2420.

Paragraph IWA-4210 was added in the Winter 1983 Addenda to require a written procedure for the storage, handling, and control of welding material. In general, this procedure would become part of the Owner's Quality Assurance Manual. The procedure would be prepared once per plant and would, like other procedures, be maintained annually thereafter. It is estimated that the time required to prepare the original procedure would be 100 hours, and the time required to maintain the procedure would be 2 hours/year/plant. Assuming that all plants presently operating and under construction would at some time implement this requirement, the total annual burden for all 119 plants, averaged over a ten-year inspection interval, would be 1428 hours/year for all plants (i.e., 100 hours/plants/10-year interval + 2 hours/year/plant x 119 plants = 1428 hours/year total for all plants).

Paragraphs IAW-6210 (a) and (c), and IWA-6340 were revised in the Winter 1983 Addenda to clarify that "inservice inspection" included preservice, as well as inservice programs, and that owners were responsible for preparing and maintaining preservice inspection plans and schedules. Preservice inspections have been a requirement of Section XI, Division 1, since its inception. Preservice inspection plans and schedules are routinely submitted to the NRC as part of the approval process for the inservice inspection program. Therefore, there is no additional recordkeeping burden associated with the revision of IWA-6210 (a) and (c), and IWA-6340.

Appendix VI, "Ultrasonic Examination of Bolts and Studs" was added to Section XI, Division 1, as a mandatory appendix in the Winter 1983 Addenda. Appendix VI provides procedures for the ultrasonic examination of Class 1 and Class 2 bolts and studs. The recordkeeping requirements of this appendix can be divided into those associated with documenting the examination procedures (i.e., initial examination development and updates) and those associated with personnel qualification records (i.e., test administration, taking tests and updating records). It is estimated that initial development of the examination procedures would require a one time effort for each plant of approximately 400 hours of a level III Examiner's time. Assuming 3 updates of these procedures during each 10-year inspection interval, it is estimated that an additional 96 hours would be expended during

each interval by Level III examination personnel for updating. The burden associated with documenting the examination procedures per plant per 10-year inspection interval would then be 496 hours. Assuming the need for three qualified ultrasonic bolting examiners per plant, and four bolting configurations for each test, it is estimated that for each inspection interval, the burden associated with personnel qualification records would be a total of 304 hours for each inspection interval for each plant (i.e., 48 hours of test administration, 160 hours of updating personnel records, and 96 hours of actual testing of Level I and II technicians). Conservatively assuming that all plants presently operating and under construction would at some time implement Appendix VI, the total annual burden of the associated recordkeeping is estimated to be 9,520 hours per year for all plants (i.e., (496 + 304 hours/plant)/(10-year interval) x 119 plants).

Paragraph IWA-2232(f) was added in the Winter 1985 Addenda to provide the conditions by which ultrasonic indications can be classified as geometric indications. The revision specifies that the maximum amplitude, location, and extent of a geometric indication shall be recorded. Since such indications are normally recorded in this manner, there is no additional paperwork burden associated with this requirement.

b. Estimated Cost Required to Respond to the Collection

Based upon the hours specified in Item 13.a, it is estimated that the cost of responding to the information collection required by the Section III, Division 1, and Section XI, Division 1, edition and addenda specified in §50.55a is a total of $1,007,216/year (1,428 + 9,520 hours x $92/hour) for 119 plants, or $8,464/year for each plant.

c. Source of Burden Data and Method for Estimating Burden

Estimates of the paperwork burden associated with implementing the identified ASME Code revisions were obtained from utility inservice inspection specialists, an inservice inspection consultant to utilities, and national lab personnel engaged in inservice inspection activities.

d. Reasonableness of Burden Estimate

The estimate of the burden is considered reasonable because of the reliable source of the burden data.

14. Reasons for Change in Burden

The estimated burden increased from 3,400 to 10,948 hours. The change in burden results from a change in ASME Code recordkeeping requirements effected by the additional addenda and edition.

15. Publication for Statistical Use

The collected information will not be published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the 10 CFR 50.55a information collection.

Attachment:
Table

Table - Summary of Burden and Cost

Recordkeeping Burden

ASME Code Section XI Reference	No. of Plants/ Year	Estimated Annual Record-keeping, Hrs/Plant	Estimated Total Annual Recordkeep-ing, Hours	Estimated Total Annual Cost to Industry, $	Estimated Total Annual Cost to Federal Govt., $
IWA-4210	119	12	1,428	131,376	nil
Appendix VI	119	80	9,520	875,840	nil
Subtotals:		92	10,948	1,007,216	nil

The Commission proposes to amend its regulations to incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, "Rules for the Construction of Nuclear Power Plant Components" and Section XI, "Rules for the Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. It is anticipated that the proposed rule will be published in the Federal Register for public comment in 1/90. Adoption of the proposed amendment would permit the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power components.

Part 4

SUPPORTING STATEMENT

for

GENERIC COMMUNICATIONS PROGRAM

10 CFR 50.71

File this to AEA 161, 182

OK-

DESCRIPTION OF THE INFORMATION COLLECTION

The generic communications program is an adjunct to the NRC regulatory oversight program and functions as an extension of the reporting requirements under 10 CFR 50.71 which require each licensee and each holder of a construction permit to maintain such records and make such reports, in connection with the licensed activity, as may be required by the conditions of the license or permit or by the rules, regulations and orders of the Commission in effectuating the purposes of the Atomic Energy Act of 1954, as amended (the Act), including Section 105 of the Act. Generic communications include bulletins, generic letters and information notices.

A. JUSTIFICATION

1. Need for the Collection of Information

The generic communications are used to disseminate information and may be used to request the submittal of information or action by addressees. They are not intended as substitutes for revised license conditions or new regulatory requirements. Most bulletins and generic letters identify the regulatory requirements that are currently contained in 10 CFR 50. Prior to proposing the bulletin or generic letter, the staff considers the potential additional burden caused by either having the NRC inspectors collect the information or having the licensees or construction permit (CP) holders provide the information in a report. Having considered both options, NRC may deem it more practical to obtain the necessary information via licensee reporting. An information notice, if used for information collection, would do so on a voluntary basis because it would be inconsistent with NRC practice to include reporting requirements in an information notice.

Proposed bulletins and generic letters that request actions or require a response are routinely reviewed by the NRC's Committee to Review Generic Requirements (CRGR), except in those rare instances where it is judged by the Director, Office of Nuclear Reactor Regulation, or the Director, Office of Nuclear Material Safety and Safeguards (NMSS), that an immediate emergency action is needed to protect the health and safety of the public. In those circumstances, no review by the CRGR is necessary and the Office Directors have the authority to issue the bulletin or generic letter.

Each proposed bulletin or generic letter to be reviewed by CRGR that does not require emergency action is categorized as either Category 1 or 2 requirements. Category 1 requirements are those which are needed to overcome problems requiring priority resolution or to comply with a legal requirement for immediate or near-term compliance.

Category 2 requirements are those which do not meet the criteria for emergency action or designation as Category 1. These are to be scrutinized carefully by the CRGR on the basis of written justification submitted by NRR or NMSS. Upon notice to the members of the CRGR, and without objection, the CRGR Chairman may exempt any Category 2 proposal from review on the grounds that he or she concludes that it involves only an insignificant effect on the NRC staff and on licensees.

The NRC believes that a reliable estimate of the annual impact of Category 1 and 2 bulletins and generic letters is possible and that this burden is logically included in 10 CFR 50.71. OK

2. Agency Use of Information

NRC periodically issues generic communications to communicate with industry on matters of generic importance or serious safety significance; i.e., if an event at one reactor raises the possibility of a generic problem, an NRC bulletin, or generic letter may be issued requesting licensees and/or CP holders to take specific actions and to submit a written report describing actions taken and other information NRC may need to assess the need for further actions to ensure public health and safety or an information notice may be submitted to voluntarily solicit information.

3. Reduction of Burden Through Information Technology

Licensees are encouraged to provide the information by the most modern technique available. There are no legal obstacles to reducing the burden associated with information collection by information notices, bulletins, and generic letters.

4. Effort to Identify Duplication

The requested information is not duplicated.

5. Effort to Use Similar Information

This information is only available from the licensees and CP holders.

6. Effort to Reduce Small Business Burden

The information notice, bulletin, and generic letter program encompassed within 10 CFR 50.73 generally does not affect small businesses. Within the last 3 years approximately 5 generic letters affected research/test reactors operated by universities.

7. Consequences of Less Frequent Collections

The information is collected on an as-needed basis to enable the NRC to resolve generic safety issues. If the Agency does not request the

information when it is needed, the health and safety of the public could be affected adversely.

8. Circumstances Which Justify Variation from OMB Guidelines

Thirty (30) days or more are allowed to respond. Therefore, this information collection does not vary from OMB guidelines.

9. Consultation Outside the Agency

When appropriate, prior to issuing a generic communication, the NRC seeks comments on the matter from industry (utilities, Nuclear Management and Resources Council, nuclear steam system suppliers, vendors, etc.). This technique has proven effective in ensuring the accuracy of statements and bringing faster and better responses from licensees.

10. Confidentiality of Information

Proprietary or confidential responses will be protected under 10 CFR 2.790 of the NRC's regulation.

11. Justification for Sensitive Questions

This information collection does not involve sensitive information.

12. Estimate of Cost to the Federal Government

Estimate of cost to the Government, which includes preparation of the bulletin or generic letter, mailing, and analysis of responses is estimated at 1,200 hours per bulletin or generic letter or 27,600 hours annually. Therefore, the total annual estimated cost to the Government is $2,539,200 (8 bulletins and 15 letters (23) annually x 1,200 = 27,600 hours; 27,600 x $92 = $2,539,200).

The NRC anticipates the preparation of one information notice annually that could involve information collection. It is estimated that approximately 160 hours of staff time will be needed to prepare such a notice. Therefore, the estimated cost could be $14,720 (160 hours x $92).

Total Federal cost for the generic communication program is, therefore, expected to be $2,553,920.

13. Estimate of Industry Burden and Cost

The number of licensees and construction permit holders affected by a particular bulletin or generic letter and the associated burden varies in each specific instance. However, an upper bound is used which assumes that all of the 119 licensees and CP holders for operating plants would respond to each of approximately 8 bulletins that is expected to be issued annually for the next 3 years. Using the historic average of 418 hours per licensee per bulletin, the annual burden for bulletins is expected to be 411,312 hours (418 x 8 = 3,344; 3,344 x 119 plants = 397,936 hours).

NRC expects to issue annually approximately 15 generic letters* containing reporting requirements. It is estimated that it would take each licensee approximately 500 hours to respond to each generic letter. This will result in approximately 892,500 hours of burden for responses (500 hours x 15 letters = 7,500 hours; 7,500 hours x 119 plants = 892,500 hours).

The NRC estimates that it will take industry approximately 40 hours per licensee to respond to an information request encompassed with an information notice. The agency estimates that approximately one notice will be issued annually. This would result in approximately 4,760 hours of burden annually (40 hours x 119 plants).

Therefore, total annual industry burden is expected to be (397,936 + 892,500 + 4,760 hours), 1,295,196 hours. Thus, the cost would be $119,158,032 (1,295,196 hours x $92).

14. Reasons for Change in Burden

The estimated increase in burden is due to the fact that during the last update review NRC's estimates were inadvertantly based on 50 utility responses; this review is based on 119 responses per plant. Moreover, experience with this program indicates that the issues are now involving more response time.

15. Publication for Statistical Use

The information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this collection of information.

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*This is in addition to the 6 generic letters estimated for 10 CFR 50.54 (f) in the Part 9 Supporting Statement. The projected burden associated with generic issues also encompasses reporting requirements pertaining to inplementation dates.

Part 5

SUPPORTING STATEMENT

10 CFR 50.48 AND APPENDIX R TO 10 CFR 50

Fire Protection

DESCRIPTION OF THE INFORMATION COLLECTION

10 CFR Part 50.48 amends the regulations to require certain provisions for fire protection in operating nuclear power plants. This action was undertaken to upgrade fire protection at nuclear power plants licensed to operate prior to January 1, 1979, by requiring resolution of certain contested generic issues in fire protection safety evaluation reports. The program on which this part is dependent is Appendix R - Fire Protection Program for Nuclear Power Facilities Operating prior to January 1, 1979, which makes requirements of certain items of fire protection guidance that have been used by the staff since the Browns-Ferry fire on March 22, 1975, to evaluate the adequacy of fire protection programs at operating nuclear power plants.

A. JUSTIFICATION

1. Need for the Collection of Information

Section 50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR 50. This fire protection plan must describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan must also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit damage to structures, systems, or components important to safety so that the capability to safely shutdown the plant is ensured. Present licensed operating plants have already met the requirement for a plan, therefore, there is no immediate burden.

Section 50.48(c)(5) requires licensees to submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and Appendix R of 10 CFR 50.

Section 50.48(c)(5) also requires licensees to submit design descriptions of modifications needed to satisfy Section III.G.3 of Appendix R to this part within 30 days after the effective date of this section and Appendix R of 10 CFR 50 (2/17/81).

Both of these requirements have already been satisfied by all licensees. Therefore, there is no additional burden.

Appendix R - Fire Protection Program for Nuclear Power Facilities Operation, requires manual fire fighting capability at each plant. It states that a fire brigade of at least five persons on each shift shall be maintained at each nuclear

power plant unit. In addition, the rule requires certain minimum levels of training for each brigade member, and training and drills for each brigade as a team. The rule also requires maintaining certain records of the training and drills provided for the brigades and brigade members. The record-keeping requirements have already been agreed to by most licensees as part of the license amendments that resulted from the staff's fire protection review of each plant. The two specific record-keeping requirements are:

A. "Section III.I.3.d."

At three-year intervals, a randomly selected unannounced drill must be critiqued by qualified individuals independent of the licensee's staff. A copy of the written report from such individuals shall be available for NRC review, and shall be retained as a record as specified in Section III.I.4 of Appendix R.

B. "Section III.I.4."

Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadening training for fire fighting within buildings shall be scheduled for all those brigade members whose performance records show deficiencies.

Description of Fire Protection Plan

These requirements will not affect nuclear power plants that were licensed to operate prior to January 1, 1979 and that already have the Appendix R requirements identified in their safety evaluation reports. 50.48(a) does not affect presently licensed plants since they have already completed these requirements with their approved fire protection programs. 50.48(a) will apply to new licensees as their applications are submitted to the NRC. No special requirement for a format or form is being imposed with this rule. Each licensee is free to develop the method and forms that best suit its individual operation. No new applications are anticipated in the next 3 years.

2. Agency Use of Information

These records are required to enable the staff to evaluate the effectiveness of each fire brigade training program and, thus, determine the expected effectiveness of each fire brigade to cope with any fire emergency which may occur.

3. Reduction of Burden Through Information Technology

There are no obstacles to reducing burden through technology.

4. Effort to Identify Duplication

There is no duplication of information.

5. Effort to Use Similar Information

This information can only be obtained from the licensee.

6. Effort to Reduce Small Business Burden

This regulation does not affect small business.

7. Consequences of Less Frequent Collection

This information is required so that the NRC can determine that fire training and drills are adequate in the event there is a fire emergency. The health and safety of the public could be affected adversely if this information is not submitted as specified.

8. Circumstances Which Justify Variation from OMB Guidelines

This regulation does not vary from OMB Guidelines.

9. Consultations Outside the NRC

Public comments were received during the proposed rulemaking stage.

10. Confidentiality of Information

Information identified as proprietary or confidential would be handled in accordance with 10 CFR 2.790. However, this information is usually not submitted as confidential.

11. Justification for Sensitive Questions

This regulation does not require sensitive information.

12. Estimate of Cost to the Federal Government

We estimate that the average review time of fire brigade drill and training records per plant is 5 staff-hours. One hundred and eleven (111) are expected to comply with this requirement annually for a total annual cost of $51,060 to the Government (111 plants x 5 staff hours/plant = 555 staff hours; 555 staff hours x $92/hr = $51,060).

13. Estimate of Industry Burden

Appendix R	No. of Respondents Affected	Hours Per Response	Annual Burden
Section III.I.3.D	111	24	2,664
Section III.1.4	111	120	13,320
		Total Annual Burden	15,984

Therefore, the estimated cost to industry is expected to be $1,470,528 ($92 x 15,984).

14. Reasons for Change in Burden

The estimated burden has changed from 15,120 to 15,984 hours because the number of plants required to submit information increased from 105 to 111 plants.

15. Publication for Statistical Use

The required information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

SUPPORTING STATEMENT
FOR
SECTION 50.54(p)
Physical Security and Safeguards Contingency Plans

DESCRIPTION OF THE INFORMATION COLLECTION

A. JUSTIFICATION

1. Need for the Collection of Information

Paragraph 50.34(c) of Title 10 of the Code of Federal Regulations requires that a physical security plan be included with each application for a license to operate a production or utilization facility. These plans are for the purpose of protection against acts of radiological sabotage and protection of special nuclear material against theft by establishment and maintenance of a physical protection system.

Section 50.34(d) of 10 CFR Part 50 specifies that each application for a license to operate a production or utilization facility shall include a licensee safeguards contingency plan in accordance with Appendix C to 10 CFR Part 73.

Section 50.54(p) of 10 CFR Part 50 requires that each licensee prepare and maintain safeguards contingency plan procedures in accordance with Appendix C of 10 CFR Part 73.

A licensee desiring to make a change which would decrease the effectiveness of a security plan prepared pursuant to Section 50.34(c), Part 73, or a licensee safeguards contingency plan (except for implementing procedures) prepared pursuant to Section 50.34(d) or Part 73, as applicable, must obtain prior approval from NRC by submitting an application for an amendment to the license pursuant to Section 50.90. For plan changes that do not descrease effectiveness, Section 50.54(p) also states that a licensee shall maintain records of such changes, which are made without prior NRC approval, for a period of 3 years from the date of the change, and shall furnish to the NRC a report containing a description of each change within 2 months after the change is made.

Additionally, Section 50.54(p) requires that the licensee review the safeguards contingency plan annually and maintain records documenting the conduct and results of the annual review along with any recommendations derived from the review. These records are to be available at the plant for inspection by NRC personnel for a period of 2 years.

2. Agency Use of the Information

Physical Security regulations include general performance requirements which recognize explicitly the need to provide protection from potential threats originating externally, from within a licensed facility, or both. The NRC staff continually reviews licensee security plans and amendments to ensure that there is comprehensive physical protection system that is capable of protecting against certain adversarial threats.

This continual review of the reactor safeguards program provides a high level of assurance to the NRC and the public that malevolent acts against operating nuclear power plants and research and test facilities, will not result in undue risk to public health and safety.

3. Reduction of Burden Through Technology

There are no legal obstacles to reducing the burden associated with this information collection.

4. Effort to Identify Duplication

This information does not duplicate nor overlap other information collections by NRC or other government agencies.

5. Effort to Use Similar Information

This information is only available from the licensee.

6. Burden on Small Business

This information collection does not affect small business.

7. Consequences of Less Frequent Collection

This information is required when an application for a license to operate a production or utilization facility is filed with NRC. There are no applications scheduled at this time. Requests for changes to current security and safeguards contingency plans are submitted on an as-needed basis.

8. Variation from OMB Guidelines

The information required under Section 50.54(p) does not vary from OMB guidelines.

9. Consultations Outside the NRC

There were no consultations during the current clearance period. During rulemaking the public and the Department of Energy were solicited for comments.

10. Confidentiality of Information

The plans get a very limited distribution and are stored in secured containers. They are protected and withheld from public disclosure pursuant to 10 CFR Part 2 (Proprietary Information), 10 CFR Part 73 (Safeguards Information), and 10 CFR Part 95 (National Security Information), as applicable.

11. Justification for Sensitive Questions

The plans are sensitive because they detail the measures and methods used to counter potential acts of sabotage and thefts of special nuclear material.

12. Estimate of the Cost to the Federal Government

The annual cost to the government is associated with analyzing and assessing the 50.54(p) amendment reports and reviews. Approximately 2.5 changes are expected annually per power plant, and .20 changes per research/test reactor. The NRC estimates that accomplishing these activities would require approximately 100 hours per power plant and 60 hours per research and test reactor. Therefore, the estimated Federal burden is expected to be as follows:

Power plants

111 x 2.5 changes/year = 278 changes
278 changes x 100 hours = 27,800 hours

Research/test reactors

48 x .20 changes = 10 changes
10 changes x 60 hours = 600 hours

Thus, total Federal burden is expected to be (27,800 + 600) 28,400 hours at the cost of $2,612,800 (28,400 hours x $92).

13. Estimate of Industry Burden

a. Number and Type of Respondents

The rule applies to each licensee who is authorized to operate a nuclear reactor or fuel reprocessing plant. There are 111 licensed nuclear power reactors and 48 research and test reactors. Thus, 159 reactor licensees are subject to the information collection requirements of 10 CFR Section 50.54(p). Currently, there are no fuel reprocessing plants in operation.

b. Estimated Hours Required to Respond to the Collection

The NRC estimates that approximately 290 50.54(p) notifications are made annually to the NRC by the licensees. It is estimated that, on the average, 200 hours are required to prepare, notify NRC, keep records, revise and file each 50.54(p) amendment for a current industry burden of 58,000 hours per year. The burden estimates were developed using a review of past 50.54(p) amendments made to the NRC by the industry. Using $92.00 per staff hour gives an industry cost of $5,336,000.

c. Reasonableness of Burden Estimates

The burden estimates were derived from consultation with licensee staff responsible for making safeguards reports and NRC staff experienced in documenting and analyzing 50.54(p) amendments.

d. Reasonableness of the Schedule for Collecting Information

The licensee has 2 months from the time of making changes that do not decrease safeguards effectiveness to report them to the NRC. This is reasonable since the time only begins to run once the changes are implemented. Licensee yearly reviews are reasonable since they generally correspond with NRC inspection

periods. Retention of the changes for 3 years is reasonable since this ensures that the information on the changes will be available for at least one inspection.

e. Method of Collecting the Information

The licensee must review the safeguards contingency plan annually and maintain records documenting the conduct and results of the annual review along with any recommendations derived from the review. The licensee may accomplish this by any procedure desired. In addition, the licensee can collect the information necessary for reporting or requesting an amendment by any method desired. The licensee must keep records of any changes and notify NRC by mail within 2 months of any changes under Section 50.54(p).

f. Record Retention Requirements

The licensee must retain records of any 50.54(p) changes for 3 years from the date of the change. The licensee must retain annual reviews of the 50.54(p) changes and recommendations that result from those reviews for a period of 2 years. This information is necessary for plant inspections by NRC personnel. Note that the burden estimate is included in 13.b.

14. Reasons for Change in Burden

The burden has increased because of the addition of research and test reactors, and the increase in licensed power plants. Therefore, the estimated notifications increased from 250 to 290.

15. Publication for Statistical Use

This information collection does not result in a publication for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 7

SUPPORTING STATEMENT

FOR

10 CFR PART 50.47, 50.54(q, r, t)
AND PART 50, APPENDIX E*

EMERGENCY PLANNING

DESCRIPTION OF THE INFORMATION COLLECTION

The Nuclear Regulatory Commission requires that all production and utilization facility licensees shall, as a condition of their license, submit emergency plans for NRC review and approval, and maintain the emergency plans up to date. Emergency plans are required to be submitted as part of the PSAR [10 CFR 50.34 (a)(10)] and FSAR or final license application [10 CFR 50.34 (b)(6)(v)] to address the elements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. In addition, copies of the detailed implementing procedures should be submitted pursuant to 10 CFR 50, Appendix E, Part V. Copies of State and local government radiological emergency response plans are also required to be submitted [10 CFR 50.33(g)].

Section 50.54(q) authorizes licensees to make changes to their emergency plans if such changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet 10 CFR Part 50. It also requires that 1 copy of these changes be sent to the appropriate NRC Regional Office and 2 copies be sent to the Document Control Desk, NRC within 30 days after the change is made. Proposed changes that decrease the effectiveness of the emergency plans are to be submitted to and approved by the Commission prior to implementation and 3 copies of such proposed changes are to be submitted.

Part 50, Appendix E, Section V requires each licensee to [illegible] to the NRC changes to emergency plan implementing procedures. One [illegible] shall be submitted to the appropriate NRC Regional Office and 2 copies [illegible] be submitted to the Document Control Desk, NRC.

Section 50.54(r) requires that each licensee who is authorized to possess and/or operate a research reactor facility under a license of the type specified in Section 50.21(c) and who had not obtained Commission approval of an emergency plan, as described in Section 50.34(b)(6)(v), prior to obtaining an operating license shall submit such a plan to the Commission for approval as part of the application for a renewal of the operating license. Each licensee who is authorized to possess and/or operate any other production or utilization facility who has not obtained Commission approval of an emergency plan, as described in Section 50.34(b)(6)(v), prior to obtaining an operating license, shall submit such a plan for approval.

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* See Supporting Statement 50.72(a), Part 10a (pages 70a-70e) for proposed rule on Emergency Response System Data.

Section 50.54(t) requires each licensee to provide for the development, revision, implementation, and maintenance of its emergency preparedness program, which shall be reviewed at least every 12 months.

A. JUSTIFICATION

1. Need for the Collection of Information

Emergency plans are needed to provide reasonable assurance that appropriate measures can and will be taken to protect the public health and safety in the event of a radiological emergency.

Changes to the emergency plans must be submitted within thirty days in order to permit the NRC to review such changes as quickly as possible. Without a quick review, changes could be made to the emergency plans which might adversely affect adequate assurance of public health and safety.

2. Agency Use of Information

The NRC must find that the emergency plans conform to the requirements of 10 CFR Part 50, and that the plans provide reasonable assurance that, in the event of an emergency, appropriate measures can and will be taken to protect public health and safety.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technology. Moreover, NRC encourages the use of such technology.

4. Effort to Identify Duplication

This information is not required by other regulations.

5. Effort to Use Similar Information

There is no source for the required information other than licensees.

6. Effort to Reduce Small Business Burden

The provisions of these regulations affect power reactors and non-power reactors (research, test and critical facilities) operated by colleges and universities. Reg. Guide 2.6 and NUREG-0849 provide information that lessens the emergency planning burden on the educational institutions.

7. Consequences of Less Frequent Collection

The NRC would not be able to determine whether plans for emergencies are adequate for assuring the health and safety of the public.

8. Circumstances Which Justify Variation From OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NRC

Efforts pertaining to emergency plans are coordinated between local, State and Federal agencies.

10. Confidentiality of Information

Proprietary, confidential or private information is handled in accordance with 10 CFR 2.790 and 10 CFR 9 of the NRC's regulations.

11. Justification for Sensitive Questions

Personal telephone numbers are needed in the event of a nuclear emergency. This information is protected in accordance with the provisions of the Privacy Act and 10 CFR 2.790.

12. Estimate of Cost to the Federal Government

NRC estimates 80 hours per year for each of the 74 nuclear power reactor sites for review of revised emergency plans and procedures. Therefore, the burden estimated for this effort is 5,920 hours. Approximately 8 hours of effort is involved for reviewing revised plans for the 48 nonpower reactors, and results in 384 hours of Federal burden. Thus, the total annual Federal burden is expected to be 6,304 hours, at a cost of $579,968 (6,304 hours x $92).

13. Estimate of Industry Burden and Cost

The burden for maintaining the emergency preparedness program is estimated to be 11,500 hours per year for each of 74 power reactor sites (851,000 hours) and 30 hours for each of 48 research/test reactor licenses (1,440 hours) for a total of 852,440 hours annually. The cost to licensees for the maintenance of their emergency preparedness program is, therefore, $78,424,480 (852,440 x $92).

14. Reasons for Change in Burden

The estimated burden hours increased from 842,250 to 852,440 hours because of the increase in the number of operating power plants.

15. Publication for Statistical Use

This information collection is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part B

SUPPORTING STATEMENT

FOR

10 CFR 50, SECTION 50.71(e)

PERIODIC UPDATE OF THE FINAL SAFETY ANALYSIS RECORD (FSAR)

DESCRIPTION OF THE INFORMATION COLLECTION

Section 50.71(e) requires each nuclear power reactor licensee to submit annually to the Commission revised Final Safety Analysis Report (FSAR) pages that reflect changes in information and analyses submitted to the Commission or prepared as a result of a Commission requirement. The updated FSAR is a reference document to be used in recurring safety analyses performed by the licensee, the Commission, and other interested parties.

The FSAR required to be updated is the original FSAR submitted as part of the application for the operating license. Supplements and amendments, submitted either in response to NRC questions or on the applicant's or licensee's own initiative following the original submittal, must be appropriately incorporated into the original FSAR to create a single, complete and integral document. This document can serve as the baseline for future changes.

A. JUSTIFICATION

1. Need for the Collection of Information

This section of the regulations is necessary because the volume of written information in the docket files of operating power reactors is large and is increasing at a rapid rate. By the time a power reactor has been in operation for a few years, much of the information in the FSAR has been modified, supplemented or superseded. This comes about by the applicant's submittal of designs and analyses supporting requested license amendments or technical specification changes, replies to regulatory requests, incident reports, and annual reports describing design and procedural changes. Consequently, it is difficult for anyone, including an NRC staff member, the licensee, or the public to be certain of the current status of a facility's design and supporting analyses.

To properly execute their respective responsibilities, the NRC staff and the licensee must work with accurate information. Problems stemming from a lack of accurate reference documents have existed for some time, but are becoming greater with the passage of time and the addition of new operating plants.

In general, the older a facility is, the more difficult it is to identify the correct information. The newly licensed facilities are not presently

a problem, but they would become so in a few years without this update procedure for licensee FSAR sets. In addition, as new staff members and licensee employees are assigned to plants with extensive licensing history and are involved in analyses and decisions affecting facility operation, the volume of reference material involved, due to lack of a single organized reference, is staggering. In such an event, the possibility of error, due to reference to outdated or incorrect material, is increased and the resultant risk to the public is likewise affected.

Paragraph 50.30(c)(2) of 10 CFR Part 50 recognizes the update need by requiring that the applicant for a construction permit update its application, which includes the Preliminary Safety Analysis Report, to eliminate superseded information and provide an index of the updated application when an Atomic Safety and Licensing Board is appointed prior to public hearing by the Atomic Safety and Licensing Board. If an operating license hearing is held, the application must be updated at that time. After the operating license is issued, various sections of Part 50 (Section 50.59, for example) require that additional safety analyses be performed for individual facility changes that affect facility safety. The present regulations reflected in 50.71(e) now require that such changes be incorporated into the FSAR.

All changes to the technical specifications are now treated as license amendments and it is appropriate to have an updated FSAR available at all times. Additionally, safety evaluations after operation of the facility has been initiated, required by proposed license amendments, technical specifications changes and other reasons, warrant at least the same supporting documentation as does the hearing process.

Section 50.71(e)(1) requires each licensee of a nuclear power plant to submit revisions containing updated information to the Commission as specified in Section 50.4, on a replacement-page basis that is accompanied by a list which identifies the current pages of the FSAR following page replacement.

Section 50.71(e)(2) specifies requirements pertaining to changes made since the previous submittal.

50.71(e)(3) specifies that a revision of the original FSAR shall be filed within 24 months of the date of the operating license, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

Section 50.71(e)(4) states that revisions shall be filed annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.

Section 50.71(e)(6) states that the updated FSAR shall be retained by the licensee until the Commission terminates the license.

2. Agency Use of Information

In addition to the uses of FSARs previously discussed, updated FSARs are currently being used for a variety of other reasons such as:

a. To evaluate proposed changes, tests or experiments made pursuant to Section 50.59 and to determine the existence of unreviewed safety questions.

b. To supply adverse operating experience to current safety reviews.

c. For operator training by licensees.

d. For project manager training, orientation, and reassignment by the Commission.

e. A reference document by management and by safety review committees.

f. By NRR and regional inspectors to assist in their facility inspections.

g. By licensing examiners to prepare exams for facility operators.

h. In planning emergency responses.

i. To evaluate operating data by NRC technical reviewers.

The NRC staff utilizes the updated information supplied by licensees in response to the reporting requirement of section 50.71(e) as a primary reference source to be employed during the numerous safety studies undertaken by licensees, the Commission, and other interested parties.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through information technologies.

4. Effort to Identify Duplication

This information is not required by any other Federal regulations.

5. Effort to Use Similar Information

The licensees are the only source for this information.

6. Effort to Reduce Small Business Burden

This information collection only involves licensees of nuclear power reactors and, therefore, does not affect small businesses.

7. Consequences of Less Frequent Collection

NRC staff members and licensee employees would not have a single organized up-to-date reference document for the plant, which could cause adverse affect on the health and safety of the public.

8. Circumstances Which Justify Variation from OMB Guidelines

The updated FSAR must be retained until the operating license is terminated because in order for the NRC to ensure the health and safety of the public at all times, the staff must be certain of the current status of a facility's design and supporting analysis.

The original and 12 copies are distributed to NRC's Central Files, Headquarters and local Public Document Rooms, the Advisory Committee on Reactor Safeguards (ACRS), consultants and various technical review and licensing staff members. These entities need copies of the voluminous updated volumes so that the official agency file unit, the public, staff members, consultants and ACRS can also be certain of current plant st[illegible].

9. Consultation Outside the NRC

None during this period.

10. Confidentiality of Information

Proprietary or confidential information is handled in accordance with 10 CFR 2.790 of the NRC's regulation.

11. Justification for Sensitive Questions

This information collection does not require sensitive information.

12. Estimate of Cost to the Federal Government

The NRC anticipates that approximately 5 staff hours will be involved annually in the handling and document control/filing systems of the updated FSAR. Thus, annual estimated cost to the Federal Government is expected to be $51,060 (5 staff hrs x 111 plants = 555 staff hours; $92/hr x 555 staff hours = $51,060).

13. Estimate of Industry Burden and Cost

Approximately 111 licensees will be affected by this reporting requirement.

The average burden per licensee for the updating is estimated to be 1,000 hours. Therefore, the annual burden for all licensees is 111,000 hours. The estimated cost to the licensees is expected to be $10,212,000 ($92 x 111,000).

14. Reasons for Burden Change

The burden changed from 105,000 to 111,000 hours because the licensees affected by this requirement increased from 105 to 111.

15. Publication for Statistical Use

The information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 9

SUPPORTING STATEMENT

FOR

SECTION 50.54(f)

COLLECTION OF INFORMATION UNDER OATH OR AFFIRMATION

DESCRIPTION OF INFORMATION COLLECTION

NRC regulations, 10 CFR Part 50, Section 50.54(f), adopted January 19, 1956 (21 FR 355), provide that the licensee upon request by the Commission, submit written statements under oath or affirmation to enable the Commission to determine whether a license should be modified, suspended, or revoked. When the staff has identified a potential health, safety, or environmental problem at a particular plant or series of plants, the staff may require the licensee or licensees to submit information to evaluate the particular situation and to make a determination whether the situation is serious enough to require that the license be modified, suspended, or revoked.

A. JUSTIFICATION

1. Need for the Collection of Information

The time allotted the licensee to respond to the request for information depends upon the perceived risk associated with the potential problem. Most responses will be requested within a 30 to 120-day period.

Periodically there are equipment failures, construction problems, and issues discovered or raised by the technical staff during the safety review and brought to the attention of the NRC through licensee reporting procedures, the safety review process itself, and by the NRC inspection staff.

Since many of the flaws and malfunctions which are detected are novel, there is little data available which would enable the NRC to predict, with certainty, what the consequences might be. To develop a reliable data base, accurately appraise the potential long-term significance of the anomaly, and determine what, if any, corrective measures may be necessary, NRC must obtain information from licensees. Should the information provided by the licensees show that there is only minor safety significance associated with the problem/situation, the facility license would not be modified, suspended, or revoked. On the other hand, the Commission may issue an Order that does modify, revoke, or suspend the license to operate a nuclear reactor.

2. Agency Use of Information

The Commission requests specific information either from one licensee, on a problem or situation believed to be unique to a particular facility, or from more than one licensee on a problem or situation believed to be generic in nature, i.e., that may affect more than one facility. Before licensees are requested to provide such information, the staff will have identified the problem or situation as one having potential safety or environmental significance.

Based on the information obtained from licensees or applicants and the staff's evaluation of the problem, new regulatory requirements may be identified. Depending upon the nature of the problem and its resolution, these new requirements could be imposed by regulation, or they could be imposed on affected facilities individually by amendment to the technical specifications or conditions of their construction permit or operating license (see 50.109, Backfitting). In addition, the NRC could issue a Regulatory Guide which would describe the nature of the problem and the method or methods found adequate by the regulatory staff for its resolution.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden associated with this information collection through information technologies.

4. Effort to Identify Duplication

This information collection is not required by any other regulations.

5. Effort to Use Similar Information

There is no source for the required information other than licensees and construction permit holders.

6. Effort to Reduce Small Business Burden

The provisions of 10 CFR 50.54(f) only affect approximately 48 universities (research/test reactors). However, a review of our records indicate that in the last three years 50.54(f) generic letters have not encompassed research/test reactors.

7. Consequences of Less Frequent Collection

Without the information provided in the licensee's written statements, timely staff action could not be taken and unsafe conditions could continue to exist, thereby potentially endangering the public health and safety.

8. Circumstances Which Justify Variation From OMB Guidelines

The requirements of Section 10 CFR 50.54(f) normally do not vary from OMB Guidelines. Only when the risk associated with a problem affects the health and safety of the public is a response requested in fewer than 30 days.

9. Consultation Outside the NRC

Occasionally NRC holds public meetings on certain issues.

10. Confidentiality of Information

Proprietary or confidential information is handled in accordance with 10 CFR 2.790.

11 Justification for Sensitive Questions

The subject regulations do not require sensitive information.

12. Estimate of Cost to the Federal Government

Prior to requesting information from the respondents, the NRC staff assesses the potential problem and identifies the needed information and how the information is to be used. This is estimated to take 200 hours for each plant-specific request and 1200 hours for each generic letter. Each specific generic letter request for information is carefully justified prior to review by the NRC Committee to Review Generic Requirements. Thus, the 5 plant specific-letters will involve approximately 1,000 hours (200 hours x 5 letters). The 6 generic letters will involve approximately 7,200 hours (1200 hours x 6 generic letters). Therefore, the estimated cost to the Federal government will be as follows:

5 plant-specific letters
1000 hours

6 generic letters [encompassing 50.54(f) provisions]
7,200 hours

Total Federal Cost
7,200 hours + 1,000 hours = 8,200 hours;
8,200 hours x $92 = $754,400

13. Estimate of Industry Burden and Cost

Plant-Specific Concern

It is estimated that perhaps as many as five plants will receive one request each year. Our estimate of the burden is that on the average each request would require several people about 2 weeks to answer. Therefore, 300 hours per request for each of 5 requests totals 1500 hours.

Generic Considerations

The number of generic letters vary and so does the number of respondents and the level of effort required to prepare the different responses. It is estimated that there will be approximately 6 letters per year requesting information pursuant to 10 CFR 50.54(f).

The six letters could involve up to 167 reactors (48 research/test reactors and a total of 119 nuclear power reactors). The burden to respond could be between 200 to 1,000 hours per letter. Most of the generic letters affect 119 power reactors and require approximately 500 hours to respond. Thus, the annual burden for power reactors would be 357,000 hours (500 x 6 letters = 3,000 hours; 3,000 hours x 119 plants = 357,000 hours). Based on current experience, no response is anticipated to be required from non-power reactors.

Total Estimated Industry Burden for plant specific and generic 50.54(f) letters would, therefore, be (1,500 + 357,000 hours) 358,500 hours; the cost would be $32,982,000 (358,500 hours x $92).

14. Reasons for Change in Burden

The annual estimate for industry burden increased from 82,500 to 358,500 hours because the generic letters discussed during the last review were estimated to involve approximately 200 hours for an average of 66 licensees. Based on experience, NRC currently estimates that approximately 500 hours will be required for 119 licensees to respond to each generic letter.

15. Publication for Statistical Use

The information collected under the provisions of 10 CFR 50.54(f) is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this collection of information.

Part 10

SUPPORTING STATEMENT

FOR

10 CFR 50.72(a)*, (b), and (c), 50.54(z)

NOTIFICATION OF SIGNIFICANT EVENTS

DESCRIPTION OF INFORMATION COLLECTION

Following the accident at Three Mile Island on March 23, 1979, the NRC staff acted to ensure the timely and accurate flow of information from licensees of operating nuclear power plants following significant events. Examples of events requiring notification:

a) Declaration of emergency situations as required by the Site Emergency Plan;

b) Any deviation from the plant's Technical Specifications;

c) Any natural phenomenon (forest fire, earthquake, tornado, hurricane) that poses a threat to the plant;

d) Injury or illness of personnel involving radioactive contamination; and

e) Initiation of a plant shutdown required by plant Technical Specifications.

These reporting requirements will affect 111 operating nuclear plants.

Dedicated telephone lines were installed at all operating power plants to facilitate direct and rapid communications between licensees and the NRC Operations Center (and Regional Offices). A line is located in each control room with provisions made for extensions to be located at other specified locations at the facility. When these phones are picked up to report significant events, they automatically ring at the NRC Operations Center and can be held open as long as needed.

A. JUSTIFICATION

1. Need for the Collection of Information

NRC issued bulletins and sent letters to each licensee asking that procedures for notification of NRC following significant events be reviewed carefully. The letters were intended to ensure that the licensees would promptly notify NRC when a reactor was determined to be in an uncontrolled or unexpected condition of operation. After this notification, a continuous communication channel was to be established and maintained between the licensee and NRC.

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* See the attached Supporting Statement (pages 70a - 70e) for the burden estimated for the proposed rule on the Emergency Response Data System.

The NRC staff evaluated licensees' responses to the letters and bulletins and determined that the reporting procedures were not providing the prompt notifications expected by the Commission. The bulletins issued to licensees did not impose reporting requirements and as a result, in several instances licensees did not notify NRC promptly. The Commission, therefore, determined that in order to protect the health and safety of the public, a rule was required. Rulemaking was initiated immediately thereafter, and resulted in an immediately final regulation (10 CFR 50.72) published in the Federal Register on February 29, 1980 (45 FR 13435). Meanwhile, the Congress provided for prompt notification in Section 201 of the Nuclear Regulatory Commission Authorization Act for Fiscal Year 1980 (Pub. L. 96-295) by amending Section 103 of the Atomic Energy Act of 1954 with a new subsection f at the end as follows: "f. Each license issued for a utilization facility under this section or Section 104 b. shall require as a condition thereof that in case of any accident which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission, the licensee shall immediately so notify the Commission. Violation of the condition prescribed by this subsection may, in the Commission's discretion, constitute grounds for license revocation. In accordance with Section 187 of this Act, the Commission shall promptly amend each license for a utilization facility issued under this section or Section 104 b. which is in effect on the date of enactment of this subsection to include the provisions required under this subsection." The final rule was published in the Federal Register on August 29, 1983, (48 FR 39045).

Section 50.54(z) requires that each licensee with a utilization facility licensed pursuant to Sections 103 or 104b of the Act shall immediately notify the NRC Operations Center of the occurrence of any event specified in 10 CFR 50.72.

The NRC staff will evaluate the information transmitted to the Commission in response to these reporting requirements and make the timely decisions required to provide adequate assurances regarding actual or potential threats to public safety.

2. Agency Use of the Information

The information collected under the provisions of the subject regulations will allow the NRC to determine when a reactor is in an uncontrolled or unexpected condition of operation. This will allow the NRC to make timely decisions following a significant event in order to protect the health and safety of the public.

3. Reduction of Burden Through Information Technology

The NRC has no objection to the use of information technologies that would reduce the burden associated with this information collection.

4. Effort to Identify Duplication

The collected information is not required by other agencies.

5. Effort to Use Similar Information

The information is only available from power reactor licensees.

6. Effort to Reduce Small Business Burden

These reporting requirements only affect nuclear power reactor licensees. Therefore, there is no burden on small businesses.

7. Consequences of Less Frequent Collection

If the NRC is not notified immediately about significant events, emergency situations would not be handled in a timely fashion. This could result in adverse affects on the health and safety of the public.

8. Circumstances Which Justify Variation from OMB Guidelines

Notification of significant events is needed in less than 30 days to ensure that plant emergencies are handled so that the health and safety of the public are not adversely affected.

9. Consultation Outside the NRC

No consultations have taken place for the final regulations during the current clearance period.

10. Confidentiality of Information

Information identified as proprietary or confidential would be handled in accordance with 10 CFR 2.790 of the NRC's regulations. However, confidential information is not anticipated.

11. Justification for Sensitive Questions

The subject regulations do not require sensitive information. However, if any reported information is within the purview of the Privacy Act, it would be handled in accordance with 10 CFR 2.790.

12. Estimate of Cost to the Federal Government

Events Analysis

The cost to the Federal government is estimated as follows:

a. Office of Nuclear Reactor Regulation - 17 person years (2,080 person hours/per year x 17 person years = 35,360 person hours)

b. Five Regional Offices - 1 person year each (2,080 person hours x 5 = 10,400 person hours)

Event Report Receipt

a. 11 persons to man the Operations Center around the clock (2,080 x 11 = 22,880 staff hours) 22,880 x $92 = $2,104,960.

b. Cost of the Emergency Notification System line for reporting events $3.5 million.

Based on the above, annual Federal cost associated with these regulations is estimated to be ((22,880 + 10,400 + 35,360 hours x $92) $6,314,800. When this is added to the Federal cost involving the emergency notification line, the total annual cost to the Federal government is expected to be approximately $9,814,800.

13. Estimate of Industry Burden and Cost

It is estimated that 40 reports annually will be received from each of 111 operating plants in response to the reporting requirement of 50.72.

The burden for each phone call is estimated to be 15 minutes. Therefore, the total annual burden for all licensees covered by this reporting requirement is estimated to be:

111 plants x 40 reports per year = 4,440 reports
4,440 reports x .25 hours = 1,110 person hours

Cost to industry is, therefore, estimated to be $102,120 (1,110 person hours x $92).

14. Reasons for Change in Burden

The estimated burden has increased from 1,050 hours to 1,110 hours because 111 plants are currently covered by the subject regulations; 105 plants were covered during the last OMB review.

15. Publication for Statistical Use

The information collected under these regulations is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 10a

SUPPORTING STATEMENT FOR

10 CFR PART 50
[(50.72(a)]

EMERGENCY RESPONSE DATA SYSTEM FOR

NUCLEAR POWER REACTOR FACILITIES

Description of the Information Collection

The proposed amendment to 10 CFR 50 would require each licensee to establish and maintain an Emergency Response Data System (ERDS) for all operating nuclear power reactor facilities except for exempt plants or those that are permanently or indefinitely shut down.

A. Justification

1. Need for the Collection of Information. Because information transmitted by voice has been found to be in error, which hampers NRC response to an emergency, the NRC regulation requires the licensee to activate the Emergency Response Data System for conditions that require the declaration of an emergency class of alert, site area emergency, or general emergency at the time that the NRC Operations Center is notified of the emergency class declaration. ERDS supplements the Emergency Notification System (ENS) for transmitting a limited set of real-time critical plant data from the licensee's on-site computer to the NRC Operations Center computer. These data are essential for NRC to fulfill its role to monitor the licensee during an alert or emergency at a nuclear power facility. In addition, the licensee would be required to maintain an ERDS configuration control program by which the NRC will be informed of any changes to the ERDS on-site hardware or software for the computer system to be assured that the effectiveness of the program is not lessened.

Section 50.72(a)(4) requires the licensee to activate the Emergency Response Data System for any condition that requires the declaration of an emergency class of alert, site area emergency, or general emergency.

Section 50, Appendix E, VI

2a requires computer systems to transmit in-plant data points for pressurized water reactors or boiling water reactors if the data points are resident in the in-plant computer.

2b requires the selected parameter sets of data to be transmitted at intervals of not less than 15 seconds or more than 60 seconds.

2c requires all link control and data transmission be established in a format compatible with the NRC receiving system.

3a requires that any hardware or software changes that affect the transmitted data points must be reported within 30 days after changes are completed.

3b requires that NRC be notified at least 30 days prior to any changes to computer hardware or software that affect transmission format and computer communication protocol.

4a requires the licensee to develop and submit an ERDS implementation plan within 75 days after the final rule is published.

2. Agency Use of Information. The real-time data transmitted utilizing the ERDS will be used by NRC to fulfill its role to monitor a licensee during an on-site alert or emergency at a nuclear power facility. In addition, information concerning any computer system hardware and software changes must be reported to the NRC to ensure system operational compatibility.

3. Reduction of Burden Through Information Technology. The result of implementing the ERDS rule will be to reduce the burden on licensees of telephonic transmission of data to the NRC during an emergency by use of a real-time data link. Information concerning the system changes are submitted infrequently under the requirements of this rule, and therefore, will not be adaptable to automated routine information technology. This information submitted also will be unique to each licensee.

4. Effort to Identify Duplication. The Information Requirements Control Automated System (IRCAS) was searched to determine duplication and none was found.

5. Effort to Use Similar Information. Presently, during an alert or higher emergency, the NRC would receive plant data from the licensee over telephone lines via the Emergency Notification System (ENS). The ERDS, which supplements the ENS, would transmit plant data in a more accurate and timely manner than the ENS, allowing more efficient and accurate assessment of emergencies to protect the public health and safety.

6. Effort to Reduce Small Business Burden. The requirements contained in this rule do not impact small business. The respondents are nuclear power plant licensees.

7. Consequences of Less Frequent Collection. Required reports are collected and evaluated on a continuing basis as events occur. The schedule for collecting the information is the minimum frequency which will permit NRC to assure that public health and safety are adequately protected.

8. Circumstances Which Justify Variation From OMB Guidelines. Contrary to the OMB Guidelines in 5 CFR 1320.6(b), certain sections of Part 50 require that licensees submit reports and transmit real-time data to the NRC.

The requirement of 50.72(a)(4) provides for electronic real-time transmittal of data to the NRC via ERDS during an alert or emergency at a nuclear power facility so that NRC has information needed to fulfill its role for protection of public health and safety.

Appendix E, Section VI, paragraphs 3a and 3b require a report within 30 days of any hardware or software changes that affect the transmitted data point identified in the Emergency Response Data System Data Point Library (data base) and changes that could affect the transmission format and communication protocol. This information is needed by the NRC to ensure that any system changes will not affect the ability to transmit critical parameters of a limited set of data to NRC so that NRC can fulfill its role to monitor a licensee during an on-site alert or emergency to protect public health and safety.

9. Consultations Outside the Agency. Information concerning ERDS was discussed at a NUMARC sponsored meeting in January 1989. It was discussed at two NRC Regulatory Information conferences in Washington, D.C. in 1989 and 1990. In addition, ERDS was discussed at an EEI Subject Area Committee meeting on Emergency Preparedness at Baltimore in September 1989.

10. Confidentiality of Information. None, except for proprietary information.

11. Sensitive Questions. None.

12. Estimated Annualized Cost to the Federal Government. The proposed rule will affect only nuclear power reactor licensees.

Cost to Government

Annualized Requirement	No. Annual Responses	Burden per Response (Man/Hr)	Total Annual Burden (Man/Hr)	Annual Cost $
50.72(a)(4) Review of Transmitted Data	10	100	1000	92,000
50,Appendix E VI 3a Review Changes Affecting Data Pts.	12	16	192	17,664
50,Appendix E VI 3b Review Changes Affecting Transmission & Protocol	12	16	192	17,664
Annual Costs After First Year	34	41+	1384	127,328
First Year Only Costs				
50,Appendix E VI 4a, Review of ERDS Implementation Plan	59*	16	944	86,848
Total First Year Cost	93	25+	2,328	214,176

13. Estimate Burden: There are 118 licensees affected by the rule change; however, only a small percentage of licensees are expected to submit a response each year. The table below reflects this.

Cost to Licensees

Annualized Requirement	No. Annual Responses	Burden per Response (Man/Hr)	Total Annual Burden (Man/Hr)	Annual Cost $
50.72(a)(4)	10	100	1000	92,000
50,Appendix E,VI 2a, 2b, & 2c	(Detail requirements of 50.72(a)(4))			
50,Appendix E VI 3a	12	12	144	13,248
50,Appendix E VI 3b	12	12	144	13,248
Annual Costs After First Year	34	38+	1288	118,496
First Year Costs 50,Appendix E VI 4a	59*	160	9440	868,480
Total First Year Cost	93	115+	10,728	986,976

* Approximately 50% of licensees have completed implementing ERDS through the voluntary program.

\+ Average burden -- Man/Hrs

14. Reason for Change in Burden. These are new requirements for which no burden has been previously assessed.

15. Publication for Statistical Use. The potential collection of information under this provision is not intended for publication for statistical use.

B. COLLECTION OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the collection of information under this provision.

Part 11

SUPPORTING STATEMENT

FOR

10 CFR 50.55(e)

REPORTING OF SIGNIFICANT DESIGN
AND CONSTRUCTION DEFICIENCIES

DESCRIPTION OF THE INFORMATION COLLECTION

Appendix B to 10 CFR Part 50 requires an applicant for a license to construct or operate a nuclear power plant to establish a quality assurance program. This program is to ensure, among other things, that all conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance, are promptly identified. Section 10 CFR 50.55(e) requires that construction permit holders report such deficiencies to the appropriate Regional Office within 24 hours. A written report is to follow within 30 days. The requirements of Section 50.55(e) were added to the regulations in 1972 to ensure that the more significant of these deficiencies be reported to the Commission.

A. JUSTIFICATION

1. Need for the Collection of Information

Without the reporting requirement of 10 CFR Section 50.55(e), the Commission would only be notified of deficiencies occurring during the design and construction of nuclear power plants through its inspection staff or through reports submitted by holders of construction permits, either voluntarily or as requested by the Commission on a case-by-case basis.

The reports submitted under Section 50.55(e) are necessary to ensure that the staff is promptly informed of deficiencies identified in design and construction so that a timely inspection and evaluation of the deficiency can be made. Timely evaluation is necessary to adequately protect public health and safety from the potential consequences of the deficiency at the plant reporting and from other similar plants, should the deficiency be generic.

2. Agency Use of Information

Specific uses made of the data reported under Section 50.55(e) include evaluation of impact of the deficiency on the quality of construction and of the adequacy of planned corrective action, identification of generic problems, planning of actions by inspection and enforcement personnel, and identification of problems in management or implementation of the quality assurance program.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through information technologies.

4. Effort to Identify Duplication

The NRC regulations contain several safety deficiency reporting requirements. Although distinctions exist between these requirements, instances have occurred where the same deficiency was evaluated and reported by different organizations. For example, a deficiency may be reported by one organization attempting to satisfy the criteria of 10 CFR Part 21 (OMB Clearance No. 3150-0035) and the same deficiency may be reported by another organization attempting to meet the differently worded criteria of 10 CFR 50.55(e). Although the intent of the regulations is for a particular deficiency to be reported under only one reporting requirement, in practice this has not always occurred.

The NRC staff is in the process of revising the current regulations to eliminate this duplicative reporting. The proposed rule was published in the Federal Register on November 4, 1988 (53 FR 44594). The intent of the proposed revisions is to establish the filing of one report for each defect or noncompliance, with the reporting obligation resting on the entity that discovers it. Although the NRC is still considering issuing the proposed rule in final form, the date of issuance is uncertain.

5. Effort to Use Similar Information

This information can only be obtained from holders of construction permits and licensees for production or utilization facilities.

6. Effort to Reduce Small Business Burden

This regulation does not affect small businesses.

7. Consequences of Less Frequent Collection

The NRC would not be informed of deficiencies or conditions adverse to quality in a timely manner.

8. Circumstances Which Justify Variation From OMB Guidelines

The 24-hour reporting requirement is necessary to inspection and Regional Office staff representatives so that a timely inspection and evaluation of the deficiency can be completed. Timely evaluation is necessary in order for the NRC to promptly ensure the protection of the health and safety of the public.

9. Consultation Outside the NRC

Public comments were received on a proposed rule change published on November 4, 1988 (53 FR 44594). Several changes have been made to the proposed rule in response to the comments.

10. Confidentiality of Information

Proprietary or Confidential Information is handled in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of Cost to the Federal Government

The NRC staff expects to receive 250 reports annually. The staff will expend approximately 8 hours to respond to each of these reports. The total burden is, therefore, estimated to be (250 x 8 hours) 2,000 hours, at a cost of $184,000 ($92 x 2,000 hours).

13. Estimate of Industry Burden and Cost

It is estimated that approximately 250 deficiency reports will be submitted annually. The burden per plant to prepare each report is estimated to be 80 hours (10 hours for each 24-hour notification report and 70 hours for each 30-day followup report). Thus, the annual burden to industry is expected to be 20,000 hours (250 x 80) at a cost of $1,840,000 ($92 x 20,000 hours).

14. Reasons for Change in Burden

The scope of 10 CFR 50.55(e) is being expanded from covering only the construction of nuclear power plants to cover all construction permits issued under 10 CFR 50.23 for any production or utilization facility. This action will provide uniform 10 CFR 50.55(e) reporting of defects and failures to comply associated with a substantial safety hazard found during construction for all the processes of design, manufacture, fabrication, installation, construction, quality assurance, testing, and inspection, which if it were to have remained uncorrected, could have adversely affected the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.

The burden has decreased because during the 1986 OMB review, 22 construction permit holders for nuclear power plants were affected by this regulation. In 1990, there were only eight construction permit holders.

15. Publication for Statistical Use

The 50.55(e) reports are not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 12

SUPPORTING STATEMENT

FOR

10 CFR 50.59(b)

REPORTS AND RECORDS FOR CHANGES, TEST AND EXPERIMENTS

DESCRIPTION OF THE INFORMATION COLLECTION

Section 50.59 of NRC regulations allows a holder of a license authorizing operation of a production or utilization facility (i) to make changes in the facility as described in the Safety Analysis Report, (ii) to make changes in procedures as described in the Safety Analysis Report, and (iii) to conduct tests or experiments not described in the Safety Analysis Report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

The licensee (111 power reactor and 48 research/test reactor licensees) shall submit a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. The report must be submitted annually or at such shorter intervals as may be specified in the license. This report generally consists of a few pages.

A. JUSTIFICATION

1. Need for the Collection of Information

The records and reports assist the NRC staff in evaluating the potential effects of 50.59 changes in relation to the health and safety of the public. The ultimate value is received in the form of ensuring the health and safety of the public.

2. Agency Use of Information

The records are used by licensees to interrelate subsequent changes and to prepare reports concerning changes, tests or experiments as required by this Section of the Regulation.

These records are also frequently used by NRC inspectors. The records provide background information needed by the NRC inspector during his or her visit to a licensed facility. The inspector uses these records to confirm the appropriateness of changes, tests or experiments, or during evaluation of abnormal occurrences.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technologies.

4. Effort to Identify Duplication

The information is not required by any other Federal regulation.

5. Effort to Use Similar Information

This information can only be obtained from licensees of operating power and nonpower (research/test) reactors.

6. Effort to Reduce Small Business Burden

The burden only affects 48 licensees (universities) of research/test reactors. This burden only occurs when licensees choose to make changes, tests, or experiments and cannot be further reduced without endangering the health and safety of the public.

7. Consequences of Less Frequent Collection

The NRC would not be able to ensure the health and safety of the public with respect to changes made to the facility without prior NRC approval.

8. Circumstances which Justify Variation from OMB Guidelines

The information required by Section 50.59 is submitted annually and, therefore, does not vary from OMB Guidelines.

9. Consultation Outside the NRC

NRC solicits comments from the public during the rulemaking phase of its final regulations. However, no changes were made to 10 CFR 50.59 during the last clearance period.

10. Confidentiality of Information

No confidential information is generally received. However, proprietary or confidential information is handled in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

This regulation does not require sensitive information.

12. Estimate of Cost to the Federal Government

It is estimated that cost to the Federal government encompasses approximately 80 hours per licensee (111 power reactor licensees; 48 research/test reactor licensees); 159 licensees x 80 = 12,720 staff hours. Therefore, the cost to the government is expected to be $1,170,240 ($92 x 12,720).

13. Estimate of Industry Burden and Cost

Estimation of Recordkeeping Requirements

Based on the staff's experience and in light of the extensive records which have to be maintained on site to meet the requirements specified in 10 CFR 50.59(b), the staff estimates that licensees for 159 facilities evaluate approximately 100 changes a year. It is also estimated that approximately 16 hours of burden each is required for records associated with the analysis of 100 changes annually. Thus, recordkeeping burden encompassed within 50.59(b) is estimated to be (1,600 hours x 159 plants) 254,400 hours. Accordingly, annual recordkeeping cost to industry will be ($92 x 254,400) $23,404,800.

Estimation of Respondent Reporting Burden

The reporting burden consist of 159 licensees submitting a summary of the changes, that have been evaluated annually. It is expected that approximately 4 hours each are required to summarize and prepare reports for approximately 100 changes per year. Thus, the reporting burden for this provision of the regulation is expected to involve 63,600 hours annually (400 x 159 plants). The annual cost to industry is, therefore, expected to be $5,851,200 (63,600 hours x $92).

Total industry burden annually would, therefore, be 318,000 hours; total annual cost would be $29,256,000 ($92 x 318,000).

14. Reasons for Change in Burden

Although the number of power reactors affected increased from 105 to 111, the estimated burden is reduced from 360,000 to 318,000 hours because during the last OMB review 75 research/test reactors were affected; 48 are currently affected.

15. Publication for Statistical Use

The collected information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in Section 50.59 information collection.

Part 13

SUPPORTING STATEMENT

FOR

10 CFR 50, APPENDIX G AND
APPENDIX H, SECTION IV; 50.60

FRACTURE TOUGHNESS TESTS, SURVEILLANCE AND REPORTS

DESCRIPTION OF THE INFORMATION COLLECTION

Appendix G to 10 CFR Part 50 specifies minimum fracture toughness requirements for the reactor coolant pressure boundary of water-cooled power reactors. Section V specifies how radiation damage to the reactor beltline is to be accounted for in the fracture control plan for the reactor. Paragraph V.C. requires that certain extra steps be taken in the event that the normal fracture analysis requirements specified in Paragraph V.B. cannot be satisfied. Paragraph V.D. requires a thermal anneal of the reactor vessel beltline if the procedures of Paragraph V.C. do not indicate the existence of an adequate safety margin. Paragraph V.E. requires that the proposed programs for satisfying the requirements of Paragraphs V.C. and V.D. be reported to the Director of Nuclear Reactor Regulation for review and approval at least three years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Paragraph V.B.

Appendix H of 10 CFR Part 50 requires a material surveillance program for each reactor vessel to monitor changes in the fracture toughness of the reactor vessel beltline materials resulting from their exposure to neutron irradiation and the thermal environment. Paragraph III requires: (a) the test results obtained from the specimens contained in each surveillance capsule shall be reported to the NRC, for each capsule withdrawal, and (b) new pressure-temperature operating limits for the reactor, based on the surveillance test results, shall be reported, if necessary. Surveillance reports provide the basis for approval of the pressure-temperature operating limits for the reactor.

Section 50.60, acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation, provisions are as follows: (a) Except as provided in paragraph (b) of 50.60, lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H. (b) Proposed alternatives to the described requirements in Appendices G and H may be approved when an exemption is granted by the Commission. In addition, the applicant must demonstrate that (1) compliance with the specified requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety, and (2) the proposed alternatives would provide an adequate level of quality and safety.

A. JUSTIFICATION

1. Need for the Collection of Information

The information in the report required by Paragraph V.E of Appendix G will be used by the staff to perform a safety evaluation of the reactor vessel. This evaluation will be the basis for approval to continue operation for a specified time and approval of the additional procedures that will be required to continue operation beyond that time. The three-year lead time is needed to provide time to obtain supplemental fracture toughness data on archive material that has been subjected to accelerated irradiation, and to evaluate the fracture analyses that will be submitted which use that data.

Section III.B contains the materials test requirements for the Charpy V-notch tests and drop weight tests. Paragraph III.B.5 specifies that records are to be kept on (1) the test results, with traceability to the material in each component, (2) the qualification of test personnel, and (3) the calibration of test equipment.

The records maintained by licensees for the life of the facility in response to the requirement are available for inspection by the staff to determine compliance with Appendix G. There is a continuing requirement that certain pieces of the data will be needed to support a licensee's fracture control plan or fracture analysis for some component in an operating plant.

The records that must be retained per Appendix G are of considerable value to the plant owner in the event of some sort of material deterioration problem or the discovery of a flaw that requires a fracture analysis. The frequency of occurrence of such situations for a given plant is difficult to estimate, but averages perhaps once every 10 years. The value to the plant owner lies in the ability to provide a sound basis for estimates of material toughness that are an essential part of the fracture analysis. However, in 1988 the staff issued Generic Letter 88-11, which requested all licensees and permittees to reassess neutron radiation embrittlement to their reactor vessels.

The impact of not obtaining the information from records would be that the fracture analyses would have to be based on conservative estimates derived from the published data base of typical material properties. The impact of an overly-conservative analysis could be the removal of some unimportant defect found in inspection with considerable economic loss due to the power outage and unnecessary exposure of maintenance personnel to radiation.

Surveillance reports required by Appendix H provide the basis for approval of the pressure temperature operating limits for the reactor. The impact of not obtaining the reports required by Paragraph IV of Appendix H would be that the pressure-temperature limits for the reactor would have to be checked against conservative estimates of radiation damage such as those given in Regulatory Guide 1.99, Revision 2. At the present time there are too many uncertainties in the assessment of radiation damage to a reactor vessel to permit a licensee to forego monitoring radiation damage and reporting the surveillance test results to the NRC.

Section 50.60 information is needed because it provides for acceptance criteria for fracture prevention measures for normal operation of light water reactors.

2. Agency Use of Information

Appendix G

The data will be used by the NRC staff in making its safety evaluation of the licensee's submittal. Material properties of the actual material in the component are an essential input to such evaluations.

Appendix H

This information is evaluated to determine whether continued operation impacts the health and safety of the public.

Section 50.60

This information is needed to ensure that the reactor vessel does not exceed radiation embrittlement limits and meets the requirements of General Design Criterion 31 and 32, as specified in Appendix A to 10 CFR Part 50.

3. Reduction of Burden Through Technology

The NRC has no objection to the use of new information technologies that could reduce the burden associated with this information collection. Moreover, use of such technology is encouraged.

4. Effort to Identify Duplication

The fracture toughness requirements stated in the subject regulations are not required by other Federal agencies.

5. Effort to Use Similar Information

There is no source for the required information other than the licensees.

6. Effort to Reduce Burden on Small Business

The subject regulations only affect nuclear power reactors and, therefore, do not involve small businesses.

7. Consequences of Less Frequent Collection

If this information were collected less frequently, the NRC would not be able to ensure that the reactor vessel has not exceeded radiation embrittlement.

8. Circumstances Which Justify Variation From OMB Guidelines

The provisions of these regulations require that this information be maintained for the life of the plant in order to detect material deteriations or flaws which might affect the health and safety of the public.

9. Consultation Outside the NRC

NRC solicits public comments during the rulemaking phase of a final rule.

10. Confidentiality of Information

Information submitted as proprietary or confidential is handled in accordance with 10 CFR 2.790(b) of the NRC's regulations.

11. Justification for Sensitive Questions

The subject regulations do not request sensitive information.

12. Estimate of Cost to the Federal Government

Appendix G

The NRC reviews annually information described below on fracture toughness. Since Appendix G reports affect the plant's licensing requirements, all of the reports must be reviewed by the NRC.

Section III.B
3 reports (1 from 3 plants) x 160 hours/report = 480 hours;
480 x $92 = $44,160

Section V.E
50 reports (1 from 50 plants) x 60 hours/report = 3,000 hours;
3,000 x $92 = $276,000

Appendix H

Since Appendix H reports are surveillance reports, the staff does a cursory review of all reports, but only reviews reports in detail when they affect licensing requirements. Hence, of the 30 reports received by the staff, only 10 get a detailed review.

10 reports x 40 hours/report = 400 hours; 400 x $92 = $36,800

Therefore, the total estimated Federal burden will be 3,880 (480 + 3,000 + 400) hours and the cost is expected to be $356,960 (3,880 x $92).

13. Estimate of Industry Burden and Cost

Appendix G

The Licensees submit the following information annually.

Section III.B
3 reports (1 from 3 plants) x 200 hours/report = 600 hours;
600 hours x $92 = $55,200.

Section V.E
50 reports (1 from 50 plants) x 80 hours/report = 4,000 hours; 4,000 hours x $92 = $368,000.

Appendix H

Thirty reports (1 from 30 plants) will be submitted annually. 30 reports x 160 hours/report = 4,800 hours; 4,800 hours x $92 = $441,660.

The total estimated annual burden for industry is, therefore, 9,400 hours at a cost of $864,800 (9,400 hours x $92).

14. Reasons for Change in Burden

The estimated burden hours have decreased from 30,820 to 9,400 hours because during the last OMB review 127 reports were anticipated under Appendix H; 30 are now expected to be submitted annually. Only 30 reports are anticipated in response to the provisions of Appendix H because we have reached the period where there is a reduction in the amount of capsules in the vessels of nuclear power plants that can be removed and tested. In addition, the following breakdown reflects specific reasons for the decrease:

1986 Review	Current Review
Section III.B 105 reports x 100 hours each	3 reports x 100 hours each
Section V.E Negligible effort	50 reports x 80 hours each
Appendix H 127 reports x 160 hours each	30 reports x 160 hours each

Burden for Section V.E has increased because the negligible burden indicated during the 1986 review was an inadvertent error. Burden for Section III.B has decreased because only 3 plants are expected to need to complete the supplemental fracture toughness tests.

15. Publication for Statistical Use

The information provided by subject regulations is not published for statistical purposes.

B. COLLECTION OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 14

SUPPORTING STATEMENT

FOR

10 CFR 50, APPENDIX J

Primary Reactor Containment Leakage Testing for
Water-Cooled Power Reactors

DESCRIPTION OF THE INFORMATION COLLECTION

10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provides for preoperational and periodic verification, by tests, of the leakage integrity of the primary reactor containment and systems and components which penetrate containment of water-cooled power reactors. Tests are conducted upon completion of construction of the primary reactor containment building (containment), periodically during each 10 year service period (approximately every 3-1/3 years) for the entire containment system, and during shutdown for refueling (approximately every 18 months but in no case at intervals greater than 2 years) for individual valves and penetrations. The reports submitted following the tests conducted every 3 1/3 years include summaries for intervening tests run at 18-month intervals. The Appendix also establishes acceptance criteria for such tests. One of the conditions of all operating licenses for water-cooled power reactors is that primary reactor containments shall meet containment leakage test requirements set forth in 10 CFR 50, Appendix J.

A. JUSTIFICATION

1. Need for the Collection of Information

Section V.B., "Inspection and Reporting of Tests," requires submission to the NRC of a summary technical report, "Reactor Containment Building Integrated Leak Rate Test," approximately 3 months after the conduct of each preoperational and periodic integrated leakage rate test, which occurs every 3 1/3 years. Furthermore, such reports must include a separate accompanying summary report analyzing and interpreting the test data for any tests that failed to meet the acceptance criteria of Appendix J. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements are also required to be included. Results of all local leakage rate tests (Type B and C) performed since the last integrated leakage rate test was performed must also be included.

The primary reactor containment is designed to contain any operational or post-accident releases of radioactivity within specified limits. Calculations of the impact of a radiological release on public health and safety are dependent upon predictable leakage from containment. The required tests ensure that the containment is built and maintained as designed, and that leakage limits are not exceeded.

2. Agency Use of the Information

Reports of preoperational leakage tests are needed by the NRC inspection and technical review/licensing staffs since these tests are the only means by which it can be verified that these structures have in fact been built within the leakage levels specified as a condition of licensing by the NRC. Information included in the report is reviewed to determine the results achieved, as well as to judge the accuracy and validity (reliability) of the data.

The reports of the periodic leakage tests are needed by the NRC in order to verify that containment leakage is maintained below the specified level throughout its operational life. Periodic information is needed for the same reasons as preoperational test information, but in addition, is compared with that in the preoperational test report and previous periodic test reports. If the preoperational or a periodic leakage test was not successfully completed, operation of the reactor would not be permitted.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technology.

4. Effort to Identify Duplication

This information is not required by other Federal regulations.

5. Effort to Use Similar Information

Power reactor licensees are the only source for this information.

6. Effort to Reduce Small Business Burden

This information collection requirement does not affect small businesses.

7. Consequences of Less Frequent Collection

The NRC would not be able to determine, in a timely fashion, whether structures have been built within limits that have been established to ensure the protection of the health and safety of the public.

8. Variation from OMB Guidelines

The provisions of Appendix J do not vary from OMB guidelines.

9. Consultation Outside of the NRC

None for the regulations currently in place. However, NRC has been considering public comments submitted on a proposed rule change that is currently being considered.

10. Confidentiality of Information

This information is usually not confidential or proprietary. If it is submitted as such, it is protected in accordance with 10 CFR Section 2.790 of the NRC's regulations.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of the Cost to the Federal Government

The NRC is expected to expend approximately one hour to review each report.

Approximately 33 reports are submitted annually. Thus, the annual cost is estimated to be $3,036 (1 hour x 33 reports = 33; 33 hours x $92 = $3,036.

13. Estimate of Industry Burden

Each licensed unit submits a report on the average of every 3 1/3 years. Out of 111 operating power reactors the NRC anticipates 33 reports annually. The burden on licensees for preparation of each report is estimated to be 366 hours. Therefore, an annual burden of 12,078 hours is anticipated (33 x 366), at an annual cost of $1,111,176 ($92 x 12,078).

A total annual burden for all licensees of 12,078 hours. Therefore, estimated industry cost is expected to be $1,111,176 ($92 x 12,078).

14. Reason for Burden Change

During the last OMB review, only 105 plants had been licensed; 111 nuclear plants are currently licensed. Therefore, the burden increased from 11,712 to 12,078 hours. (However, because of a miscalculation, the 1986 clearance package indicated the burden was 25,620 hours.)

15. Publication for Statistical Use

The information collected under Appendix J is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 15

SUPPORTING STATEMENT

FOR

10 CFR 50.35(b) - PERIODIC RESEARCH AND DEVELOPMENT REPORTS

DESCRIPTION OF THE INFORMATION COLLECTION

Section 50.35(b) specifies that "The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions".

A. JUSTIFICATION

1. Need for Collection of Information

The reports required under 10 CFR 50.35(b) would keep the staff apprised of the progress and findings of licensee research and development programs and increase the likelihood that any safety problems would be resolved in a timely manner.

2. Agency Use of Information

The staff will review information submitted in accordance with 10 CFR 50.35(b) and evaluate the results of research and development programs. This evaluation is to determine what, if any, corrective measures were appropriate and develop regulatory procedures including revisions to existing review processes and possible facility modification, if necessary. This procedure allows the Commission, by special reference in a facility construction permit, to request information concerning ongoing research and development activities that are in support of a construction permit.

This reporting requirement has not resulted in the submittal of any information from licensees over the last 3 years. NRC requests renewal of the clearance for this section, however, in order to receive timely information from licensees on potential new technological developments for both power reactor and fuel reprocessing systems should they occur. Ongoing research and development programs throughout the industry create the possibility of safety-related issues arising at any time. The NRC staff must be able to obtain information from licensees concerning current research projects in order to make informed judgements about the effects of current research on future licensing actions.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technologies, and the NRC encourages the use of such technologies.

4. Effort to Identify Duplication

This provision is not required by other Federal regulations.

5. Effort to Use Similar Information

Licensees for nuclear power plants are the only source for this information.

6. Effort to Reduce Small Business Burden

This provision only affects licensees for nuclear power plants and, therefore, does not affect small business.

7. Consequences of Less Frequent Collection

Less frequent collection could mean that research information that could impact future licensing actions might not be available on a timely basis.

8. Circumstances Which Justify Variation from OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NRC

None.

10. Confidentiality of Information

Confidential or proprietary information is treated in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

This provision does not request sensitive information.

12. Estimate of Cost to the Federal Government

NRC does not anticipate any responses from industry based on this regulation. Therefore, there is no anticipated cost to the government for the next 3 years.

13. Estimate of Industry Burden and Cost

There is no anticipated response from industry for the next 3 years.

14. Reasons for Change in Burden

There is no change in the burden since the last OMB review.

15. Publication for Statistical Use

The information collected under these provisions is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in these information collections.

SUPPORTING STATEMENT

FOR

10 CFR 50.34(g) - GUIDANCE FOR IMPLEMENTATION Part 15a

OF THE STANDARD REVIEW RULE

NUREG-0906

DESCRIPTION OF THE INFORMATION COLLECTION

The NRC conducts a detailed review of all applications for licenses to construct and operate nuclear facilities. In March 1982, the NRC adopted a final rule, 50.34(g), which requires the applicants for a construction permit (CP), operating license (OL), preliminary design approval (PDA), or final design approval (FDA) provide, as part of the material currently required by 10 CFR 50.34, an evaluation of the differences from the Standard Review Plan (SRP) (NUREG-0800) acceptance criteria, for those applications docketed after the effective date of this regulation.

A. JUSTIFICATION

1. Need for Collection of Information

The objective of the requirement contained in 10 CFR 50.34(g) and of the implementing guidance of NUREG-0906 is to allow the limited NRC staff resources to quickly focus on those areas involving differences from the SRP acceptance criteria in order to make the most effective use of the staff's resources. Experience has shown that such differences usually involve issues of safety significance and require the greatest amount of time to resolve. Since the applicants are familiar with their plant's designs, they are in a better position to identify the differences from the SRP acceptance criteria during the normal course of preparing the technical supporting information for an application.

2. Agency Use of Information

The Standard Review Plan (SRP) reflects the NRC's detailed interpretations of the acceptable means to satisfy the applicable regulatory requirements, which ensure that the proposed facilities can be constructed and operated without any undue risk to the health and safety of the public. Because of limited resources, the NRC staff conducts audit reviews of the Safety Analysis Reports (SARs) submitted in accordance with an application, in accordance with the review procedures in the SRP.

The material currently found in SARs does not lend itself to ready identification of the differences from the SRP acceptance criteria. These differences are often found in responses to staff questions or

during meeting discussions. Differences from the SRP acceptance criteria do not necessarily imply nonconformance with regulatory requirements. However, they do reflect a departure from accepted practice that should be highlighted by the licensee to ensure a thorough staff review.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through information technologies.

4. Effort to Identify Duplication

This provision is not required by other Federal regulations.

5. Effort to Use Similar Information

Licensees for nuclear power plants are the only source for this information.

6. Effort to Reduce Small Business Burden

This provision only affects licensees for nuclear power plants and, therefore, does not affect small business.

7. Consequences of Less Frequent Collection

Less frequent collection could impact NRC's detailed interpretations of the acceptable means to satisfy the applicable regulatory requirements which ensure that a proposed facility can be constructed and operated without any undue risk to the health and safety of the public.

8. Circumstances Which Justify Variation from OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NRC

None.

10. Confidentiality of Information

Confidential or proprietary information is treated in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

This provision does not request sensitive information.

12. Estimate of Annual Cost to the Federal Government

One design review which differs from the SRP is expected to be submitted during this clearance period.

The staff estimates that it will take approximately 1,000 staff hours to review the CE 80+ design. Therefore, the estimated annual burden is expected to involve approximately 333 hours, at an annual cost of $30,636 (333 hours x $92).

13. Estimate of Annual Burden and Cost to Industry

It is estimated that approximately 3,000 burden hours will be needed by the licensee (Combustion Engineering) to complete a total review required by 10 CFR 50.34(g) for the CE 80+ design.

Therefore, the annual burden is expected to be 1,000 hours at a cost of $92,000 (1,000 hours x $92).

14. Reasons for Change in Burden

During the previous OMB review, the NRC did not anticipate a submittal that would be required to meet the provisions of the subject regulation. However, in early 1991 the NRC expects the licensee for CE 80+ to meet the requirements stated in 10 CFR 50.34(g).

15. Publication for Statistical Use

The information collected under this provision is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 16

SUPPORTING STATEMENT

FOR

10 CFR 50.71(b) and APPENDIX C

ANNUAL FINANCIAL REPORT
AND
FINANCIAL REQUIREMENTS

DESCRIPTION OF THE INFORMATION COLLECTION

The requirement for the annual financial report, including the certified financial statements, arises from the Atomic Energy Act of 1954, as amended, Section 182 "License Applications." Section 182(a) provides, among other things, that each application or a license shall state such information as the Commission, by rule or regulation, may determine to be necessary to decide the financial qualifications of the applicant as the Commission may deem appropriate for the license.

This reporting requirement affects 119 licensees (111 operating licensees and 8 construction permit holders).

A. JUSTIFICATION

1. Need for the Collection of Information

Section 10 CFR 50.71(b) provides for the filing of annual financial reports, including certified financial statements, of facility licensees with the Commission. The fundamental purpose of the financial qualifications provision is the protection of the public health and safety and the common defense and security. A licensee's financial qualifications may affect his ability to meet his responsibilities on safety matters.

The Commission reserves the right to require additional financial information during the operation of a facility, particularly in cases in which the nuclear power plant will be commonly owned by two or more existing companies, or in which financing depends upon long-term arrangements for the sharing of the electric power output of the facility by two or more electric power generating companies. The annual financial report provides updated financial information after a construction permit has been issued for a nuclear power plant.

2. Agency Use of Information

The annual financial reports, and any other pertinent material that may be needed, are used by NRC staff for financial monitoring of the respondents. If it appears that any respondent is experiencing financial difficulties,

this information is used for management consideration of any appropriate actions.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through the use of information technologies.

4. Effort to Identify Duplication

This information requested in Section 50.71(b) and Appendix C is not required by any other regulation. The financial information required by Section 50.33(f) for applications for construction permits and operating licenses is used to establish financial qualifications needed before NRC can approve the applications and is not duplicated here (see Part 1 of the Supporting Statement package).

5. Effort to Use Similar Information

There is no source for the required information other than licensees.

6. Effort to Reduce Small Business Burden

This information collection only affects power reactor licensees and, therefore, does not affect small businesses.

7. Consequences of Less Frequent Collection

If the information is not submitted when required, there could be a situation where a licensee's financial qualifications are questionable, which could affect the licensee's ability to meet responsibilities on safety matters.

8. Circumstances Which Justify Variation from OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NPC

None during this updating period.

10. Confidentiality of Information

Confidential financial information is protected in accordance with 10 CFR 2.790 of the NRC's regulations.

11. Justification for Sensitive Questions

This information collection does not require sensitive information.

12. Estimate of Cost to the Federal Government

Responses are required from 119 licensees. It is estimated that approximately one hour of staff effort is required to review each of the submittals. Therefore, total cost to the Federal government is expected to be ($92 x 119) $10,948.

13. Estimate of Industry Burden and Cost

The staff estimates that approximately one hour is needed by industry to respond to these requirements. Therefore, that would involve 119 hours of industry burden hours at a cost of ($92 x 119 hours) $10,948.

14. Reasons for Burden Change

The decrease from 127 to 119 hours is consistent with the decrease of the number of plants that are currently affected since the last OMB review.

15. Publication for Statistical Use

The collected information is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 17

SUPPORTING STATEMENT
FOR
PROPERTY DAMAGE/ACCIDENT RECOVERY
INSURANCE

10 CFR 50.54(w)(3)
AND FINAL RULE 10 CFR 50.54(w)(4)(i)(ii)

DESCRIPTION OF THE INFORMATION

Section 10 CFR 50.54(w) requires that each electric utility licensee under 10 CFR Part 50 for a production or utilization facility shall take steps to obtain onsite property damage insurance available at reasonable costs and on reasonable terms from private sources or to demonstrate that it possesses an equivalent amount of protection. Proceeds from such insurance will be used, in the event of an accident, to stabilize and decontaminate the reactor to prevent a situation that could threaten public health and safety. Under § 50.54(w)(2) [changed to § 50.54(w)(3)], lead reactor licensees (approximately 55) are required to report annually on the amount and sources of this required insurance. Under § 50.54(w)(3) [changed to § 50.54(w)(4)(i)(ii)], a licensee suffering an accident is required to submit a cleanup plan outlining the steps and costs needed to complete decontamination and cleanup and to allow release of the remaining insurance proceeds for non-cleanup purposes.

Section 50.54(w)(4)(i) will adopt proposals submitted in petitions for rulemaking by reactor licensees and their representatives to establish a threshold of $100 million before a cleanup plan would be required. Section 50.54(w)(4)(ii) will require licensees to inform the Director of the Office of Nuclear Reactor Regulation in writing when the reactor is and can be maintained in a safe and stable condition so as to prevent any significant risk to the public health and safety. Within 30 days after the licensee informs the Director that the reactor is in this condition, or at such earlier time as the licensee may elect or the Director may for good cause direct, the licensee shall prepare and submit a cleanup plan for the Director's approval. The cleanup plan must identify and contain an estimate of the cost of each cleanup operation that will be required to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the NRC for authority to decommission the reactor and to surrender the license voluntarily.

A. JUSTIFICATION

1. Need for the Collection of Information

Licensees of commercial nuclear power plants are required to submit annually proof that they carry onsite property damage/accident recovery insurance available from private sources. A licensee suffering an accident is also required to submit a cleanup plan within 30 days after the reactor is stabilized. This cleanup plan will also explicitly include, under the final rule, costs of performing each cleanup operation. This information is required to demonstrate that licensees are complying with NRC's requirement to carry adequate accident recovery insurance and, in

the event of a reactor accident, to provide the NRC with sufficient information to monitor cleanup and to allow insurance proceeds to be released from the decontamination priority and to be used for non-cleanup purposes.

2. Agency Use of Information

The information submitted by licensees is used by the NRC staff to ensure that licensees are complying with the requirements to maintain appropriate levels of onsite property damage/accident recovery insurance and to use the proceeds from this insurance for decontamination and cleanup after an accident before any other purpose.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reduce burden associated with this collection of information through technology. Moreover, NRC encourages the use of such technology.

4. Effort to Identify Duplication

This information is not duplicated by any other agency.

5. Effort to Use Similar Information

This information is only available from power reactor licensees.

6. Effort to Reduce Small Business Burden

This information collection requirement only affects power reactor licensees.

7. Consequences of Less Frequent Collection

Annual reporting of coverage is considered the least frequent reporting interval which will still give reasonable assurance of insurance coverage in order to protect the health and safety of the public in case of an accident.

8. Variation from OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NRC

None during this period.

10. Confidentiality of Information

The NRC does not anticipate the receipt of confidential information. However, if confidential information is submitted, it would be protected in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

These regulations do not require sensitive information.

12. Estimate of Cost to the Federal Government

Total staff review time per year for the annual report is 15 minutes/licensee x 55 licensees = 16 staff hours. Given the assumption of salary of $92.00 per hour, the total dollar cost to the Federal government is expected to be $1.472 annually (16 hours x $92). The cleanup plan required to be submitted by a licensee suffering an accident is expected to require approximately 1,000 staff hours, or $92,000 per review (1,000 hours x $92). However, it is unlikely that there will be an accident of the severity addressed in §50.54(w). Thus, the NRC estimates no burden for this potential reporting requirement.

13. Estimate of Industry Burden and Cost

Average reporting burden to each licensee for the annual report is a letter to NRC of usually no more than one paragraph indicating both the amount of onsite property damage insurance being carried by the licensee and the insurer(s) from whom the insurance was obtained. Time to complete this is estimated to be no greater than 4 hours per licensee. No significant variation in burden among licensees is expected. There are currently 55 licensees affected by the reporting requirements. Thus, the current annual burden is no more than 220 hours. The estimated industry cost is, therefore, $20,240 ($92.00 x 220). Because an accident requiring a licensee to submit a cleanup plan is unlikely, no burden for this requirement is projected. It is estimated that a licensee required to prepare and submit a cleanup plan after an accident* could face a burden of $184,000 (2,000 hours x $92.00).

14. Reasons for Change in Burden

There has not been a burden change since OMB review of the information collections for the final rule.

15. Publication for Statistical Use

There are no plans to publish this information for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

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*If there is such an accident of the severity that is specified in Section 50.54(w)(4).

Part 18

SUPPORTING STATEMENT

FOR

HYDROGEN CONTROL REQUIREMENTS

10 CFR 50.44(c)

DESCRIPTION OF THE INFORMATION

Information required by 10 CFR Section 50.44(c) is associated with the final rulemaking (December 2, 1981, 46 FR 58484) on hydrogen control requirements related to inerting of Mark I and II boiling water reactors, hydrogen recombiner capability and high point vents. This rule was revised on January 25, 1985 (50 FR 3498) to include boiling water reactors with Mark III containments and pressurized water reactors with ice condenser containments.

The requirements to submit analyses for both the hydrogen control system and the demonstration of survivability during a hydrogen burn would apply to Mark III BWRs and ice condenser PWRs in various stages of the licensing process. Due to the similarities between plants, it is estimated that seven reports will be received from power reactor licensees for seven plants (Milestone 1, Dresden 2 and 3, Quad Cities 1 and 2, Cooper and Oyster Creek).

A. JUSTIFICATION

1. Need for the Collection of Information

The accident at Three Mile Island, Unit 2 (TMI-2) resulted in a severely damaged or degraded reactor core, a concomitant release of radioactive material to the primary coolant system, and a fuel cladding-water reaction which resulted in the generation of a large amount of hydrogen. The Nuclear Regulatory Commission has taken numerous actions to correct the design and operational limitations revealed by the accident. Included in these actions are several rulemaking proceedings intended to improve the hydrogen control capability of light-water nuclear power reactors. On October 2, 1980, the Nuclear Regulatory Commission published in the Federal Register (45 FR 65466) a notice of proposed rulemaking on "Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations" (Interim Rule). The notice concerned proposed amendments to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to improve hydrogen management in light-water reactor facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core.

On March 23, 1981, the Commission published in the Federal Register (46 FR 18045) a notice of proposed rulemaking on "Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." The notice proposed a set of licensing requirements applicable to construction permit applications that stemmed from lessons learned from the TMI-2 accident.

On May 13, 1981, the Commission published in the Federal Register (46 FR 26491) a notice of proposed rulemaking on "Licensing Requirements for Pending Operating License Applications" (OL Rule).

As a follow-up to the October 2, 1980 notice of proposed rulemaking, the Commission published a notice of final rulemaking on December 2, 1981 (46 FR 58484) on hydrogen control requirements related to inerting of Mark I and II boiling water reactors, hydrogen recombiner capability and high point vents.

The Commission has also considered the ability of all non-inerted light-water nuclear power reactors, particularly pressurized light-water reactor facilities with ice condenser type containments and boiling light-water reactor facilities with Mark III type containments, to withstand an accident with the concomitant generation of large amounts of hydrogen, such as the type which occurred at Three Mile Island, Unit 2 (TMI-2). As a result, three new amendments to the regulations were proposed for public comment on December 23, 1981 (46 FR 62231).

Hydrogen control requirements related to boiling water reactors with Mark III containments and pressurized water reactors with ice condenser containments were issued as final rule revisions to 10 CFR 50.44 on January 25, 1985 (50 FR 3498). That rule revision requires: (a) that ice condenser and Mark III plants be provided with a hydrogen control system capable of accommodating the hydrogen generated by the metal water reaction involving up to 75 percent of the fuel cladding; (b) that those reactors not relying upon an inerted atmosphere for hydrogen control show that certain important safety systems must be able to function during and following hydrogen burning; and finally (c) analyses to be submitted to justify the hydrogen control systems selected and to provide assurance that containment structural integrity will be maintained and important safety systems will continue to function following a hydrogen burn, for those plants in (a) and (b) above.

2. Agency Use of Information

The information contained in the analyses described in Item A.1 is necessary to permit the NRC staff to perform an evaluation to determine if the requirements for hydrogen control and safety equipment functioning during a hydrogen burn are met. Without this information the NRC staff could not evaluate the design of the hydrogen control systems selected or determine whether or not needed safety equipment could indeed function during a hydrogen burn.

3. Reduction of Burden Through Technology

There are no legal obstacles to reducing the burden associated with this information collection through technology.

4. Effort to Identify Duplication

There is no duplication involved with this information collection.

5. Effort to Use Similar Information

The required information is only available from the licensees.

6. Effort to Reduce Burden on Small Business

The provisions of 10 CFR 50.44(c) do not affect small business.

7. Consequences of Less Frequent Collection

Less frequent collection could result in the NRC not determining whether or not safety equipment could function during a hydrogen burn.

8. Variation from OMB Guidelines

This information collection does not vary from OMB Guidelines.

9. Consultation Outside the NRC

Comments from the public were considered during the rulemaking phase.

10. Confidentiality of Information

Any information identified as proprietary or confidential will be protected in accordance with the provisions of 10 CFR 2.790.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of Cost to the Federal Government

The evaluation of the reports by the NRC staff will require approximately 960 hours for each Mark III BWR and ice condenser PWR for a total of 6,720 staff hours and a total cost of $618,240 (7 plants x 960 x $92).

13. Estimate of Industry Burden

The reporting of the design and survivability analyses for the seven plants (four Mark III BWRs and three ice condenser PWRs) will require approximately 1,500 hours per plant for a total of 10,500 burden hours. Therefore, total cost to industry is expected to be $966,000 ($92 x 10,500 hours).

14. Reasons for Change in Burden

NRC did not receive the reports projected during the last OMB review. Consequently, the estimated burden has increased from six to seven plants as it includes the six reports not received during the last previous clearance period.

15. Publication for Statistical Use

The information collected under the subject regulation is not published for statistical purposes.

B. COLLECTION OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 19

SUPPORTING STATEMENT

FOR

10 CFR 50 SECTION 50.49

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

DESCRIPTION OF THE INFORMATION COLLECTION

The information collection requirements contained in 10 CFR 50.49 consist of the following:

A. 50.49(d): establishment of records listing all electrical equipment covered by the rule, its performance characteristics, its electrical characteristics, and the environmental conditions in which it must operate.

B. 50.49(f): requirement of methods to be followed for qualifying electric equipment important to safety. The analysis required for this provision of the regulations must be maintained at the site of the plant.

C. 50.49(g): identification of the electrical equipment already qualified prior to the effective date of the rule and submission of a schedule for qualifying or replacing the remaining electrical equipment.

D. 50.49(h): notification of any significant equipment qualification problems that may require extension of the completion date within 60 days of its discovery.

E. 50.49(i): submission of an analysis by an applicant for an operating license to ensure that the plant can be safely operated pending completion of the environmental qualification of electrical equipment.

F. 50.49(j): maintenance of records of electrical equipment qualified under these regulations, retained for the entire period during which the item is installed or stored for future use. These records must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section:

(1) Is qualified for its application; and

(2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

These records must be maintained by the applicant or licensee for meeting the requirements specified in 50.49(d). This regulation requires the applicant or licensee to prepare a list of electric equipment important to safety, and to include the following information for electric equipment important to safety in a qualification file:

(1) The performance specifications under conditions existing during and following design basis accidents.

(2) The voltage, frequency, load, and other electrical characteristics for which the performance specified.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform.

A. JUSTIFICATION

1. Need for Collection of Information

Nuclear power plant equipment important to safety must be able to perform its safety functions throughout its installed life. The regulations are designed to ensure the NRC that the electrical equipment will be able to perform its accident mitigation functions under the postulated environmental conditions. To accomplish this objective, the regulations require licensees and applicants to qualify the essential electrical equipment. Qualification methods include testing as the primary method and analysis in combination with partial type test data or operating experience.

By its Memorandum and Order CLI-80-21, dated May 23, 1980, the Commission directed that the "DOR* Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," and NUREG-0588, "Interim Staff Position of Environmental Qualification of Safety-Related Electrical Equipment," form the basis for the requirements licensees and applicants, respectively, must meet for environmental qualification of electrical equipment. This Memorandum and Order also included certain reporting and recordkeeping requirements with which licensees of the operating nuclear power plants are required to comply. The recordkeeping requirements, in general terms, are contained in Sections XI and XVII of 10 CFR 50, Appendix B. This regulation codifies the Commission's current requirements for the qualification of electrical equipment and explicitly states the reporting and recordkeeping requirements.

2. Agency Use of Information

The reports and records required by 10 CFR 50.49 allow NRC to determine whether 119 licensees (111 operating plants; 8 plants under construction) meet requirements pertaining to environmental qualification of electrical equipment.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this regulation through information technologies.

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*Division of Operating Reactors.

4. Effort to Identify Duplication

These requirements are not duplicated in other Federal regulations.

5. Effort to Use Similar Information

This information can only be obtained from licensees of nuclear power plants.

6. Effort to Reduce Small Business Burden

This requirement only affects nuclear power reactor licensees and, therefore, does not affect small businesses.

7. Consequences of Less Frequent Collection

The provisions of 10 CFR Section 50.49 require the licensee to set up a program for the environmental qualification (EQ) of electric equipment. Therefore, this regulation involves a one-time effort.

8. Circumstances Which Justify Variation from OMB Guidelines

The records discussed in 10 CFR 50.49 are required to be maintained for the life of the component so that the NRC can determine if equipment important to safety at nuclear power plants meets specified performance requirements.

9. Consultation Outside the NRC

NRC Staff participates in the development of national IEEE standards. Since 1971 these IEEE standards have included specific requirements for qualification methods. Some of these methods have been adopted by the NRC.

10. Confidentiality of Information

Confidential or Proprietary Information is handled in accordance with 10 CFR 2.790 of the NRC's regulations.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of the Cost to the Federal Government

See the enclosed Table.

13. Estimate of Industry Burden and Cost

See the enclosed Table.

14. Reasons for Change in Burden

During the last OMB review 22 construction permit holders were required to meet the provisions of 10 CFR 50.49. Only eight plants are currently under construction. Thus, the estimated burden for construction permit holders has been reduced from 46,540 to 27,900 hours. Total burden for operating licenses has increased by 600 hours as 111 plants are currently licensed. Only 105 were licensed during the last clearance renewal.

15. Publication for Statistical Use

This information collection is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used for this information collection.

Enclosure:

Table - Estimate of Burden (Industry and Federal)

10 CFR 50.49

Table

Estimate of Burden

	Reporting requirements	Annual Compliance Burden: For 111 operating nuclear power plants (h/plant)		For 8 plants under construction (h/plant)	
		Licensees	Govt.	Applicants	Govt.
50.49(d)	Development of list of electrical equipment and its characteristics (one time only)	Completed	Completed	2000	40
50.49(g)	Submission of a schedule for qualification and replacement (one time only)	Completed	Completed	N/A	N/A
50.49(h)	Reporting of significant qualification problem (Average 2 responses annually per plant)	20	4	20	4
50.49(i)	Submission of a safety analysis report (one time only)	Completed	Completed	N/A	N/A
	Sub Total Licensee/Applicant Burden	20 h/plant		2,020 h/plant	
	Total Licensee/Applicant Burden:	20 h/plant x 111 plants =		2,020 h/plant x 8 plants =	
	2,220 + 16,160 = 18,380 hours	2,220 h		16,160 h	
	Sub Total Burden to Government		4 h/plant		44 h/plant
	Total Burden to Government:		(4 h/plants x 111 plants) =		(44 h/plant x 8 plants =
	444 + 352 = 796		444 h		352 h

Estimates of Cost to Industry for Reporting

The total cost to industry is estimated to be (18,380 hr x $92) $1,690,960.

Estimates of Cost to Federal Government for Reporting

The total cost to the Government is estimated to be (796 hr x $92) $73,232.

10 CFR 50.49 Table

Estimate of Burden

		Annual Compliance Burden			
	Reporting requirements	For 111 operating nuclear power plants (h/plant)		For 8 plants under construction (h/plant)	
		Licensees	Govt.	Applicants	Govt.
50.49(d)	Development of list of electrical equipment and its characteristics (one time only)	Completed	Completed	2000	40
50.49(g)	Submission of a schedule for qualification and replacement (one time only)	Completed	Completed	N/A	N/A
50.49(h)	Reporting of significant qualification problem (Average 2 responses annually per plant)	20	4	20	4
50.49(i)	One time submission of a safety analysis report	Revision	Revision	N/A	N/A
	Sub Total Licensee/Applicant Burden	20 h/plant		2,020 h/plant	
	Total Licensee/Applicant Burden:	20 h/plant x 111 plants =		2,020 h/plant x 8 plants =	
	2,220 + 16,160 = 18,380 hours	2,220 h		16,160 h	
	Sub Total Burden to Government		4 h/plant		44 h/plant
	Total Burden to Government:		(4 h/plants x 111 plants) =		(44 h/plant x 8 plants =
	444 + 352 = 796		444 h		352 h

Estimates of Cost to Industry for Reporting

The total cost to industry is estimated to be (18,380 hr x $92) $1,690,960.

Estimates of Cost to Federal Government for Reporting

The total cost to the Government is estimated to be (796 hr x $92) $73,232.

Table

Estimate of Burden

		Annual Compliance Burden			
	Recordkeeping requirements	For 111 operating nuclear power plants (h/plant)		For 8 plants under construction	
		Licensees	Govt.	Applicants	Govt.
50.49(f) and (j)	Maintain records which demonstrate qualification	80	N/A	80	N/A
	Sub Total Licensee/Applicant Burden	80 h/plant		80 h/plant	
	Total Licensee/Applicant Burden:	80 h/plant x 111 plant =		80 h/plant x 8 plants =	
	8,880 + 640 = 9,520	8,880	Total	640 h	
	Sub Total Burden to Government		N/A		N/A
	Total Burden to Government:		N/A		N/A

None

Estimates of Cost to Industry for Recordkeeping

The total annual cost to industry is estimated to be (9,520 x $92) $875,840.

Estimates of Cost to Federal Government for Recordkeeping

The total annual cost to the Government will be negligible as records are inspected as part of the routine inspection effort.

Total Burden and Cost to Industry: 27,900 hours; $2,566,800

Total Burden and Cost to Government: 796 hours; $73,232

Part 20

SUPPORTING STATEMENT

FOR

ANTICIPATED TRANSIENT WITHOUT SCRAM

10 CFR 50.62

DESCRIPTION OF THE INFORMATION COLLECTION

Section 50.62 requires the installation of certain equipment in nuclear power plants, in order to prevent and mitigate anticipated transient without scram (ATWS) events. The licensee for a nuclear power plant is required, by 10 CFR 50.62(c)(6), to submit a copy of the design and installation plans to the NRC to ensure that the design and installation of the equipment will perform its intended safety function.

In addition, 10 CFR 50.62(d) requires the licensee to submit a schedule to the NRC for implementing the requirements of the rule. This provision allows the establishment of implementation schedules that are tailored to the safety priority needs and resources of the individual licensee.

A. JUSTIFICATION

1. Need for the Collection of Information

An ATWS is an expected operational transient (such as a loss of feedwater, loss of condenser, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor trip system (RTS) to shut down the reactor. The RTS consists of those power sources, sensors, initiation circuits, logic matrices, bypasses, circuit breakers, interlocks, racks, panels and control boards, and actuation and actuated devices, that are required to initiate reactor shutdown, and includes the control rods and control rod mechanisms as well. That portion of the RTS exclusive of the control rods and control rod mechanisms is referred to as the scram system. ATWS is a cause of concern because under certain postulated conditions they could lead to severe core damage and release of radioactivity to the environment. The ATWS question involves safe shutdown of the reactor during a transient, if there is a failure of the RTS. There have been precursors to an ATWS such as the failure of the automatic portion of the RTS at the Salem 1 nuclear generating station on February 25, 1983, although manual shutdown was accomplished after 30 seconds, and no core damage or release of radioactivity occurred. Section 50.62 requires improvements in the design and operation of nuclear power plants to reduce the likelihood of failure of the reactor protection system to shut down the reactor following anticipated transients, and to mitigate the consequences of anticipated transients without scram events. This will significantly reduce the risks of nuclear power plant operation.

2. Agency Use of the Information

The NRC will review the design and installation plans to ensure that the equipment will perform its intended safety function.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection through the use of information technologies.

4. Effort to Identify Duplication

The provisions of these regulations are not duplicated by other government regulations.

5. Effort to Use Similar Information

Reactor power licensees are the only source [illegible] information.

6. Effort to Reduce Small Business Burden

The provisions of the subject regulation do [illegible]ct small businesses.

7. Consequences of Less Frequent Collection

The NRC would not be able to ensure in a timely fashion that the ATWS equipment will perform its intended function to reduce the risks of nuclear power plant operation, and thereby protect the health and safety of the public.

8. Circumstances Which Justify Variation from OMB Guidelines

The information collection involving the subject regulation does not vary from OMB Guidelines.

9. Consultations Outside the NRC

There were no consultations in the current clearance period.

10. Confidentiality of Information

Proprietary or confidential information is protected in accordance with 10 CFR 2.790 of the NRC regulations.

11. Justification for Sensitive Questions

This regulation does not require sensitive information.

12. Estimate of Cost to Federal Government

Watts Bar was the last plant that needed to meet the provisions of this regulation. That plant's submittal has been reviewed. Therefore, the only effort anticipated during this OMB review period is to (1) issue a generic

letter that will handle the matter pertaining to 19 BWR plants that need to replace trip units to satisfy diversity requirements; (2) complete staff reviews related to changes to the Standard Technical Specifications; and (3) complete inspections covered under 10 CFR 50.70. With respect to item (1) the projected cost is covered under 10 CFR 50.71 and item (2) Technical Specification efforts are covered under 10 CFR 50.36.

13. Estimate of Industry Burden and Cost

There is no burden anticipated under 50.62 during this clearance period. The burden anticipated for the licensees to respond to the generic letter pertaining to BWR plants replacing trip units to satisfy diversity requirements is encompassed within the burden discussed in the Part 4 Supporting Statement for Generic Communications, 10 CFR 50.71.

14. Reasons for Change in Burden

The burden projected during our last update was 6,604 hours because 127 plants were expected to respond. All the plants have now responded.

15. Publication for Statistical Use

The collected information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the subject information collection.

Part 21

SUPPORTING STATEMENT FOR 10 CFR 50.61

"FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS"

DESCRIPTION OF THE INFORMATION COLLECTION

Pressurized thermal shock events are system transients in pressurized water reactors that can cause severe overcooling followed by immediate repressurization to a high level. The thermal stresses caused by rapid cooling of the reactor vessel inside surface combine with the pressure stresses to increase the potential for fracture if an initiating flaw is present in low toughness material. Such material may exist in the reactor vessel beltline, adjacent to the core, where neutron radiation gradually embrittles the material during plant lifetime. The toughness of reactor vessel materials is characterized by a "reference temperature for nil ductility transition" (RT_{NDT}). The value of RT_{NDT} at a given time in a vessel's life is used in fracture mechanics calculations to determine whether assumed pre-existing flaws would propagate as cracks when the vessel is stressed.

The Pressurized Thermal Shock (PTS) rule, 10 CFR § 50.61, adopted on July 23, 1985, establishes a screening criterion, a limiting level of embrittlement beyond which operation cannot continue without further plant specific evaluation. The screening criterion is given in terms of RT_{NDT}, calculated as a function of the copper and nickel contents of the material and the neutron fluence according to the procedure given in the PTS rule, and called RT_{PTS} to distinguish it from other procedures for calculating RT_{NDT}.

The PTS rule [paragraph (b)(1)] requires each PWR licensee to report the results of the calculations of predicted RT_{PTS} values for each beltline material, (including the copper, nickel and fluence values that provided the basis for the calculations) from the time the licensee submits the report up to the expiration date of the operating license (EOL). The rule further provides [(paragraph (b)(3))] that if RT_{PTS} for the controlling material is predicted to exceed the screening criterion before EOL, the licensee should submit plans and a schedule for such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion.

Paragraph (b)(3) further provides that any information believed to improve the accuracy of the RT_{PTS} values significantly shall be reported to the NRC.

Finally, paragraph (b)(4) of the present PTS rule requires licensees of plants that would reach the screening criterion before EOL despite the flux reduction program to submit a plant-specific safety analysis justifying operation beyond the screening criterion. The licensee must submit the analysis at least 3 years before the plant is predicted to reach that limit. Regulatory

Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" provides guidance for the preparation of the report and describes acceptance criteria that the NRC staff would use.

In response to the PTS rule, the licensees of operating PWRs have submitted the fluence predictions and chemical composition data and (with 2 or 3 exceptions) these have now been accepted. A number of licensees have undertaken flux reduction programs for those plants having high values of RT_{PTS}. Some of these are still under review. The earliest date for submittal of requests to operate beyond the screening criterion [per paragraph (b)(4)], is expected to be in the late 1990's. A proposed rule was published in the Federal Register in December 1989, which requires resubmittal of the RT_{PTS} analysis due to changes in the procedure for calculating RT_{PTS}. The final rule is expected to be published in March 1991. The identity of the licensees who make submittals may change, and the number of licensees affected by paragraph (b)(4) may increase to as many as 7 to 9, because more plants will probably reach the screening criterion before end of life when the amended rule is in effect, which is anticipated in March 1991.

A. JUSTIFICATION

1. Need for the Collection of Information

The primary purpose of the amended rule is to change the procedure for calculating RT_{PTS} to reflect recent findings that embrittlement is occurring faster than predicted by the PTS rule for some reactor vessel materials. Paragraph (b)(1) of the amended rule requires all licensees of operating PWR's to resubmit projected values of RT_{PTS} using the new procedure. If the copper and nickel contents and fluence projections are the same as in the previous submittal, they need only be listed; but if there are changes, justification must be provided. If a licensee has already submitted any part of the information required by paragraph (b)(1) of the amended rule, the licensee may simply reference the earlier submittal.

However, each PWR licensee will be required to prepare a submittal, because the amended rule to paragraph (b)(1) requires that projected values of RT_{PTS} be submitted for specific dates: (1) the time of submittal, (2) the expiration date of the operating license, (3) the projected expiration date if a change in the operating license has been requested (e.g., a change from 40 years after the construction permit was received to 40 years after the operating license was issued), and (4) the projected expiration date of a renewal term if a request for license renewal has been submitted. Fluence estimates will be required for these dates in order to calculate RT_{PTS} for these dates, and the fluence estimates must be obtained from the licensee to make certain that the licensee's latest plans for future core loading are included. Thus, the estimate of reporting burden of the amended rule contains the estimate for a submittal of a letter report from each of the 73 licensees.

Paragraph (b)(3) of the amended rule requires submittals from some of the PWR licensees. Those whose calculations pursuant paragraph (b)(1) and

(b)(2) reveal that they may not exceed the screening criterion before end of licensed life, or the end of a license renewal term, are to submit plans for a flux reduction program if further flux reduction is practicable. (Many have already done some flux reduction in response to the 1985 PTS rule.) An upper-bound estimate of the number of affected licensees is 22 plants, if all of those who would reach the screening criterion before the end of a 20-year license renewal period choose to do additional flux reduction.

The new request for materials information (RT_{PTS} values) contained in this amended rule is required because: the calculation of RT_{PTS} involves fluence projection for a different location in the vessel wall and for different times in the life of the vessel than do the calculations for pressure-temperature limits. Moreover, the calculation of RT_{PTS} is a prelude to the preparation of a flux reduction plan to avoid reaching the screening criterion, as required by the amended rule.

2. Agency Use of Information

The information and analyses will be reported on the plant's docket pursuant to the provisions of 10 CFR 50.4. The Project Manager (PM) will coordinate review of the information and analyses by the appropriate branch (depending upon technical subjects covered) leading to coordinated NRC staff action to ensure the requirements of the regulation are met.

There is a safety issue involved in the information collection requirements described above. By reviewing the submittals from the PWR licensees the NRC can make certain that (a) all of them are aware of the potential threat to the integrity of their reactor vessel from pressurized thermal shock events, and (b) those that need to consider additional flux reduction in order to stay below the screening criterion will become aware of the need as early as possible, when it will do the most good.

3. Reduction of Burden Through Information Technology

The information required is a one-time effort that does not lend itself to the use of new information technologies, but the NRC encourages the use of any technology that would reduce the burden on the licensee.

4. Effort to Identify Duplication

There are no other NRC requirements regarding analyses for flux reduction or plant PTS safety analyses. However, materials information leading to calculation of an RT_{NDT} value for the reactor vessel is submitted in response to the requirements of Appendices G and H, 10 CFR Part 50 (See Supporting Statement included in this submittal as Part 13.) For new plants, it appears in the final safety analysis report. During the operating life, the information is updated by the individual plant submittals that support requests for changes in the pressure-temperature limits given in Technical Specifications.

5. Effort to Use Similar Information

If a licensee has already submitted the information required by paragraph (b)(1) of this proposed amendment, the licensee may simply reference the earlier submittal.

6. Effort to Reduce Small Business Burden

This rule applies only to licensees of pressurized water reactors (commercial power plants), none of which are small businesses.

7. Consequences of Less Frequent Collection

This rule require one-time information collection only.

8. Circumstances Which Justify Variations From OMB Guidelines

There are no variations from OMB Guidelines in this collection of information.

9. Consultations Outside NRC

NRC has reviewed the overall PTS recommendations on several occasions with the Advisory Committee on Reactor Safeguards (ACRS), including the information gathering aspects. The ACRS was in basic agreement with NRC's recommendations (letter to Victor Stello, Executive Director for Operations, NRC, from Forrest J. Remick, Chairman, ACRS, April 11, 1989).

10. Confidentiality of Information

All reports submitted will be subject to public disclosure. Proprietary information (information of commercial value or "trade secrets") could be withheld, if properly identified by the licensee at the time of submitting the report, in accordance with the provisions of 10 CFR 2.790.

11. Justification of Sensitive Questions

No sensitive information is requested under these regulations.

12. Estimate of Cost to Federal Government

The submittals by the licensee will be evaluated by the staff, at the estimated cost given below.

1) RT_{PTS} assessment

The staff estimates that reevaluations of RT_{PTS} values will be submitted by 73 licensees 6 months after the effective date of the amendment to the regulation.

The submittals will be evaluated by the Materials and Chemical Engineering Branch and the Reactor Systems Branch of the Office of Nuclear Reactor Regulation. It is estimated that 10 hours will be

required to review each submittal. Total review time is estimated at 730 staff hours at an estimated cost of $67,160 (73 x 10 hours x $92).

2) It is estimated that an analysis and schedule for implementation of a flux reduction program will be submitted by 22 licensees nine months after the effective date of the regulation.

The submittals will be reviewed and evaluated by the Reactor Systems Branch. It is estimated that 25 hours will be required to review each submittal. Total review time is estimated to be 550 staff hours at a cost of $50,600 (22 x 25 hours x $92). The expenditure will be made in FY-91.

Total Annual Federal Cost (averaged over 3 years) = $39,253.

13. Estimate of Burden and Cost to Industry

Number and Type of Respondents

The licensees of all PWR plants would be subject to the regulation.

It is estimated that seventy-three plants would be affected by the RT_{PTS} assessment; and approximately twenty-two plants would also be affected by the flux reduction analyses.

Estimated Licensee Burden

1) RT_{PTS} assessment 120 staff hours per plant - (73 x 120 = 8,760 staff hours total)

2) Flux reduction analyses - 700 staff hours per plant - (700 x 22 = 15,400 staff hours total)

3) The NRC does not anticipate that a licensee will need to meet the provisions of paragraph (b)(3) during the time this clearance is in place.

Summing these two requirements, the total estimated burden hours will be 24,160 in the first year after publication of the PTS rule. All burden is accumulated in the first year. Of the 73 licensees, the reporting burden will be 120 hours each for an estimated 51 of them and 820 hours each for the 22 licensees, who must do both analyses, for a total of 24,160 hours.

Total Estimated Annual Burden (averaged over three years) = 8,050 hours. The annual estimated cost would, therefore, be $740,600 (8,050 hours x $92).

The basic information is available to each licensee through ongoing reactor vessel integrity and surveillance programs and previous responses to the PTS rule. The method for estimating is based on engineering judgment by the NRC staff and its understanding of the assessment of the integrity of the vessel.

The estimates given above represent the best judgment of the NRC staff, and are based on actual experience with the responses to the earlier PTS rule.

14. Reasons for Change in Burden

The burden increased because there are more PWR licensees licensed since the 1986 Part 50 OMB submittal, and because of the new screening criterion required by 50.61(b)(4).

15. Publication for Statistical Use

The information collected under the provisions of 10 CFR 50.61 is not published for statistical purposes.

16. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this information collection.

Part 22

OMB SUPPORTING STATEMENT

10 CFR 50.64

Limiting the Use of Highly Enriched Uranium
in Research Reactors

DESCRIPTION OF THE INFORMATION COLLECTION

The regulations encompassed within 10 CFR 50.64 are to limit the use of highly enriched uranium (HEU) fuel in research and test reactors (nuclear non-power reactors). Moreover, these regulations require that new non-power reactors use low enriched uranium (LEU) fuel and that existing non-power reactors replace HEU fuel with LEU fuel when available.

A letter from licensees with supporting documentation and analyses, or a proposed schedule is the method of transmitting the required information that will allow careful and complete review.

A. JUSTIFICATION

1. Need for the Collection of Information

A Commission policy statement published August 24, 1982 (47 FR 37007), explains NRC's interest in reducing the use of HEU in research reactors. This interest stems from NRC's licensing responsibility for both domestic use and for export of HEU and concern about risks of theft or diversion of this material.

The policy statement also describes a continuing program to develop and demonstrate the technology that will facilitate the use of reduced enrichment fuels. The reduced enrichment for research and test reactors (RERTR) program was initiated by the Department of Energy (DOE) and is managed by the Argonne National Laboratory. Its objective is to prove the ability of new LEU fuels to replace existing HEU fuel without significant changes to existing reactor cores or facilities, or significant decrease in performance characteristics of the reactors.

Information shows that a major consideration is the cost of conversion. NRC shares the licensees' expressed view that conversion costs should largely or entirely be financed by the Federal government. Historically, the DOE and its predecessor agencies have provided significant support to research and test reactor programs. The availability of Federal support will be considered in determining the availability of LEU fuel and final schedules for conversion.

A regulation entitled "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Non-power Reactors," 10 CFR 50.64, was published in the Federal Register on February 25, 1986 (51 FR 6519). This regulation is intended only to reduce the risk of theft or diversion of HEU fuel used in

non-power reactors. The reduction in domestic use of HEU fuel may encourage similar action by foreign research reactor operators, and thereby reduce the amount of HEU fuel in international use.

Non-power reactors are required to use LEU fuel or use HEU fuel of enrichment as close to 20 percent as is available and acceptable to the Commission. Section 50.64(c)(1) of the rule states that any request with supporting documentation for a determination that a reactor has a unique purpose was to be submitted by September 29, 1986. Section 50.64(c)(2) of the rule requires each non-power reactor licensee authorized to possess and use HEU fuel to develop and submit, at 12-month intervals after March 27, 1987, to the NRC's Director of the Office of Nuclear Reactor Regulation a proposed schedule for conversion to LEU fuel or to use HEU fuel as close to 20 percent as is available and acceptable to the Commission. This proposed schedule will be based upon the availability of replacement fuel acceptable to the NRC and consideration of other factors such as the availability of shipping casks, financial support, and reactor usage. A final schedule will then be determined by the Director.

Section 50.64(c)(2)(iii) requires supportive safety analyses so as to meet the schedule established for conversion.

2. Agency Use of Information

A respondent is required to submit a request with supporting information pursuant to 10 CFR 50.64(c)(1) to the NRC. The NRC will use the information to make a determination that the nuclear non-power reactor has a unique purpose as defined in 10 CFR 50.2.

A respondent will develop and submit to the NRC pursuant to 10 CFR 50.64(c)(2) a proposed schedule for meeting the requirements of 10 CFR 50.64(b)(2) or (3). The proposed schedule must be based upon availability of replacement fuel acceptable to the Commission and consideration of other factors such as the availability of shipping casks, financial support, and reactor usage. NRC will use the proposed schedule plus the results of the successful accomplishment of the tasks set out in DOE's RERTR program and the development of commercially available replacement fuel to determine a final schedule.

The proposed schedule for meeting the requirements of 10 CFR 50.64(c)(2) will require a comparison between the licensee's existing fuel design and fuels developed or projected for development under the documented RERTR program. Coordination with NRC to formulate proposed schedules for regulatory review and with DOE to develop fuel procurement and supporting equipment schedules will be required. Twelve months is considered a reasonable time for development of the proposed schedule.

NRC will review the supportive safety analyses required by the provisions of Section 50.64(c)(2)(iii). Subsequent to this review, the Director of the Office of Nuclear Reactor Regulation will issue an appropriate enforcement order directing both the conversion and, to the extent consistent with protection of the public health and safety, any necessary changes to the license, facility, or procedures.

3. Reduction of Burden Through Information Technology

The NRC has no objection to the use of new information technologies and generally encourages its use.

4. Effort to Identify Duplication

The information required by this regulation is not duplicative of other required information.

5. Effort to Use Similar Information

This information is only available from non-power reactor licensees.

6. Effort to Reduce Small Business Burden

This information collection affects colleges and universities. The schedules for conversion of fuel are necessary so that the NRC can ensure proper controls pertaining to risks of theft or diversion of HEU.

7. Consequences of Less Frequent Collection

Information to justify use of HEU or to schedule its discontinuance is necessary to protect the health and safety of the public.

8. Variation from OMB Guidelines

This information collection does not vary from OMB guidelines.

9. Consultation Outside NRC

None for the final rule currently in place. During the development of the proposed rule, NRC considered extensive comments from the U.S. State Department, DOE, and the non-power reactor owners. Implementation of the final rule required extensive coordination between NRC, DOE, and the affected licensees.

10. Confidentiality of Information

Information identified as confidential or proprietary is handled in accordance with the provisions specified in 10 CFR 2.790.

11. Justification for Sensitive Questions

This regulation does not request sensitive information.

12. Estimate of Annual Cost to the Federal Government

Section 50.64(c)(1). NRC staff time for making a determination for each of the two "unique purpose" reactor requests will require approximately 120 hours. The total staff time for two requests would be 240 hours.

Section 50.64(c)(2). NRC staff time for consideration of a schedule proposed by a non-power reactor licensee and determination of a final

schedule will require approximately 120 hours for each of approximately 12 licensees (7 who have not submitted schedules and 5 who have submitted schedules) for a total of 1,440 hours. The total burden would, therefore, be 1,680 hours (240 + 1,440 hours). Thus, the real burden to the Federal government is expected to be 560 hours (1,680 ÷ 3 years the clearance will be in place), at an annual cost of $51,520 (560 x $92).

In addition, there is a total contract cost of $35,000 for each of the 12 reviews. Thus, the total contract expense is expected to be $420,000; the annual contract cost while this clearance is in place would, therefore, be approximately $140,000 ($420,000 ÷ 3 years).

Annual cost to the Federal government is, therefore, expected to be $191,520 ($51,520 and $140,000 for contract cost).

13. Estimate of Annual Industry Burden and Cost

The NRC anticipates 7 more respondents to submit on a one-time basis a proposed schedule for conversion to LEU fuel or for use of HEU fuel of enrichment as close to 20 percent as is available and acceptable to the Commission [§50.64(c)(2)]. Two licensees have already submitted unique purpose exemptions in accordance with §50.64(c)(1). These requests and proposed schedules will be submitted once during the facility operating lifetime prior to meeting the requirements in 10 CFR 50.64(c)(2). The records resulting from 50.64(c)(2)(iii) have a retention period that is specified in the licensee's Technical Specifications.

Burden estimates discussed in a and b below are based on discussions with NRC staff who have been through the licensing process.

a. Section 50.64(c)(1). Approximately 300 hours each are required for two respondents to develop the request with supporting documentation for a "unique purpose" determination to be submitted to the Director of the Office of Nuclear Reactor Regulation. Burden hours would, therefore, be 600.

b. Section 50.64(c)(2). Approximately 120 hours each are required for approximately seven respondents to develop the proposed schedule and submit the proposed schedule to NRC. This burden will be approximately 840 hours (7 x 120 hours). In addition, it is anticipated that approximately 1,000 hours will be expended by each licensee to prepare appropriate safety analyses as specified in Section 50.64(c)(2)(iii). Therefore, approximately 7,000 hours (1,000 hours x 7 licensees) would be expended for this effort.

Therefore, the total annual burden to industry is expected to be 2,813 hours (600 + 840 + 7,000 hours divided by a 3-year clearance period), at an annual cost of $258,796 (2,813 hours x $92).

14. Reasons for Change in Burden

The estimated burden is reduced from 4,920 hours to 2,713 hours because the requirements under 50.64(c)(1) are more than two-thirds complete, and because only seven respondents are expected to comply with the requirements

of 50.64(c)(2) versus the 31 expected during the 1986 review. Moreover, NRC inadvertently omitted estimated burden for industry to prepare the safety analysis required in 50.64(c)(2) during the 1986 review.

15. Publication for Statistical Use

The collected information is not published for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the collection of information.

OMB SUPPORTING STATEMENT FOR THE
THE ECCS RULE CONTAINED IN APPENDIX K
AND SECTION 50.46 OF 10 CFR PART 50

DESCRIPTION OF THE INFORMATION COLLECTION

The revised Emergency Core Cooling System (ECCS) Rule affects the information collection requirements presented in 10 CFR Part 50 in the following manner:

1. The rule provides an alternate method of meeting the requirements contained in Appendix K and Section 50.46 of 10 CFR Part 50. It permits licensees or applicants to analyze ECCS performance using realistic calculations. This method of calculation may remove some operating restrictions and, thus, motivate licensees to submit realistic analyses for review. The previous rule did not have this feature; therefore, this aspect of the effective rule represents a voluntary information collection burden to the industry. Realistic analyses are not required of licensees not electing this option.

2. 10 CFR Part 50, Section 50.46(a)(3)(i) and (ii), respectively, require:

   (i) Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

   (ii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in 10 CFR 50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with §50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC-approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the

criteria set forth in paragraph (b) of this section is a reportable event as described in 10 CFR 50.55(e), 50.72 and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with §50.46 requirements.

3. Appendix K.II.1.b of 10 CFR Part 50 requires that a complete listing of each computer program be furnished to the NRC upon request in the same form as used in the evaluation model (EM). Therefore, the computer listings for each modified EM exceeding 50°F limit stated earlier will no longer be mandatory; but submitted only when NRC determines that such information is needed to perform its review. NRC does not anticipate the need to request such information during this clearance period.

A. JUSTIFICATION

1. Need for the Collection of Information. The regulations in 10 CFR Part 50 are promulgated by the NRC pursuant to the Atomic Energy Act of 1954, as amended, (68 Stat. 919) and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242) to provide for the licensing of production and utilization facilities. In order to determine licensee compliance with the regulations set forth in 10 CFR 50.46 and Appendix K of 10 CFR Part 50, the NRC needs to know what models and methods have been used to assess ECCS performance.

2. Agency Use of Information. The information identified earlier will be used to determine licensee compliance with the requirements of Appendix K and §50.46(b) of 10 CFR Part 50 and, thus, ensure that the reactor operates within the limits required to protect the public health and safety. If not in compliance, the information will allow NRR to assess how and when compliance to the applicable requirements will be achieved.

   Without the information required in Section II of Appendix K, the NRC staff would be unable to determine the adequacy of the calculational methods used to evaluate ECCS performance.

3. Reduction of Burden Through Information Technology. There are no legal obstacles to the reduction of information collection requirements through information technology. However, the limited frequency of submittal makes it impractical to reduce the burden through the use of technology.

4. Effort to Identify Duplication. The Information Requirements Control Automated System was searched to determine duplication, and none was found.

5. Effort to Use Similar Information. No similar information is currently available.

6. Effort to Reduce Small Business Burden. The rule does not affect small businesses.

7. Consequence of Less Frequent Collection. The frequency with which this information is collected is determined by how often the accepted ECCS EM

is modified and whether these changes significantly affect the calculated peak clad temperature. Less frequent collection could adversely affect the public health and safety.

8. Circumstances Which Justify Variation From OMB Guidelines. This information collection does not vary from OMB Guidelines.

9. Consultations Outside the NRC. None during this update review.

10. Confidentiality of Information. The NRC will protect classified, proprietary and sensitive information according to the guidelines provided in 10 CFR 2.790.

11. Justification for Sensitive Questions. No sensitive information is requested.

12. Estimated Annualized Cost to the Federal Government. It is expected that three generic calculations using realistic models will be submitted during the clearance period, and three modified EM models will be submitted during the next 5-year period, or 6 submittals per year. Staff review of a modified EM will require one half of a staff year, and a generic analysis of ECCS performance will require an average of one staff-year per submittal. The number of reviews performed per year as a result of this regulation is estimated as follows:

Modified EM Submittals:	0.6/yr
Generic Model Submittals:	1.0/yr
Total:	1.6/yr

The annualized cost to the NRC would be $192,004 (2,087 hours × $92) for the generic analyses and $96,002 (1,044 hours × $92) for modified EM submittals. The total annualized cost to the NRC for both generic and modified submittals is estimated as $288,006.

The rule requires that a schedule for completing the actions needed to comply with applicable Appendix K and § 50.46(b) requirements be submitted to NRC with each analysis. Schedule review would require 4 hours of staff time per submittal. At $92 per hour and 1.6 submittals per year, the annualized cost to the NRC would be $589 (1.6 × 4 hours = 6.4; 6.4 × $92).

The annual reports required by the provisions of Section 50.46(a)(3)(ii) will result in negligible burden. One report is expected to be submitted by 111 licenses. It is estimated that it would only take approximately 15 minutes for the staff to peruse these reports.

The total cost to the NRC is, therefore, $288,595 ($288,006 + $589) annually.

13. Estimate of Industry Burden.

The annual burden to industry for modified EM submittals, realistic generic model submittals, schedule and computer printout submittals is estimated at 4,008 burden hours. Attachment A provides a breakdown of this burden.

This is based on an estimate that the average cost to industry for performing an analysis of ECCS performance is 2,500 person hours, a modified EM will involve 1,500 hours, and that preparation and submittal of a schedule would involve 8 person hours; an EM printout is expected to involve approximately one hour. Based on an estimate of an average of 1.6 submittals annually (one generic realistic model submittal and 0.6 modified EM submittals annually), the total burden to industry is estimated at 4,008 person hours annually.

One annual report required by Section 50.46(a)(3)(ii) will be submitted by 111 licensees. Based on the staff's experience, it is estimated that it will take each licensee approximately 4 hours to prepare each report. This would result in approximately 444 hours annually.

Therefore, the total annual burden for industry is estimated to be 4,452 hours (4,008 + 444), at an annual cost of $409,584 (4,452 hours x $92).

14. Reasons for Change in Burden. The slight burden decrease is because during the previous clearance period it was estimated that it required 13 hours to prepare a schedule submittal. Based on current experience, it is estimated that only 8 hours are required.

15. Publication for Statistical Use. The information being collected is not expected to be published for statistical use.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the collection of information.

Enclosure:
Attachment A

ATTACHMENT A

OMB STATEMENT FOR THE ECCS RULE CONTAINED
IN APPENDIX K AND SECTION 50.46 OF 10 CFR PART 50
ANNUAL BURDEN TO INDUSTRY

	Responses per year	Hours per response	Total annual burden hours	Estimated annual industry cost
1. Section 50.46 Requirements				
- Realistic EM Submittals	1	2500	2500	$230,000
- Modified EM Submittals	0.6	2500	1500	$138,000
- Schedule Submittal	1	8	8	$736
- EM Printout Submittal	0	0	0	0
2. Appendix K. II.1.b.	0	0	0	0
3. Reports under 50.46(a)(3)ii)	111	4	444	$40,848
Totals:	113.6		4,452	$409,584

Part 24

SUPPORTING STATEMENT
FOR
STATION BLACKOUT
SECTION 10 CFR 50.63

DESCRIPTION OF THE INFORMATION COLLECTION

The provisions of 10 CFR 50.63 require each licensed light-water-cooled nuclear power plant to be able to withstand for a specified duration and recover from a station* blackout as defined in § 50.2. The capability for coping with station blackout of specified duration shall be determined by an appropriate coping analysis or by meeting the requirements of § 50.63(c)(2). Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review. Licensees are required to submit the following information to NRC: (1) a proposed station blackout duration to be used in determining compliance with the provisions of 10 CFR 50.63, including a justification for the selection based on factors identified in § 50.63, including a justification for the selection based on factors identified in § 50.63(a) and set forth below; (2) a description of the procedures that will be implemented for station blackout events for the duration determined in (1), above, and for recovery therefrom; and (3) a list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of § 50.63, for the specified station blackout duration determined in (1), above, and a proposed schedule for implementing the stated modifications. Licensees shall use the following factors in determining the specified station blackout duration: (1) the redundancy of the onsite emergency AC power sources; (2) the reliability of the onsite emergency AC power sources; (3) the expected frequency of loss of offsite power; and (4) the probable time needed to restore offsite power.

A. JUSTIFICATION

1. Need for the Collection of Information

This issue concerns the reliability of the alternating current (AC) electrical power for essential and nonessential service in nuclear power plants. Normal AC electrical power is supplied primarily by the onsite/offsite (preferred) power supply; redundant onsite emergency AC power systems also are provided in the event that the preferred power source is lost. The loss of both the preferred and onsite emergency AC power systems results in a condition called station blackout.

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*A station can consist of 2 or 3 units (or plants). For example, Millstone Station, Units 1, 2 and 3, encompasses 3 plants.

The AC electrical power systems provide power for various safety systems including reactor core decay heat removal and containment heat removal. These systems are essential for preserving the integrity of the reactor core and the containment building. The reactor core decay heat also can be removed for a limited time period by safety systems that are independent of AC power. If a total loss of all AC electrical power persists for a sufficient time that the capability of the AC-independent system to remove decay heat is exceeded, core melt and containment failure could result.

This issue has been studied extensively by the Commission under Unresolved Safety Issue A-44, Station Blackout. As a consequence of these studies, the NRC amended its regulations by adding a new § 50.63 to 10 CFR to require that light water reactor nuclear power plants be designed to withstand a total loss of AC electrical power for a specified time duration and maintain reactor core cooling during that period. This requirement is intended to provide further assurance that a station blackout will not adversely affect the public health and safety. The requirements and information collection contained in § 50.63, "Loss of All Alternating Current Power," are as follows:

(a) Requirements. Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2 in accordance with the provisions of 10 CFR 50.63.

(b) Implementation - Information Submittal.

For each light-water-cooled nuclear power plant licensed to operate on or before July 21, 1988, the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by April 17, 1989. For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment [July 21, 1988], the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with § 50.63, including a justification for the selection based on the four factors identified in paragraph § 50.63(a);

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in § 50.63(c)(1)(i) and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of § 50.63(a), for the specified station blackout duration determined in § 50.63(c)(1)(i), and a proposed schedule for implementing the stated modifications.

After consideration of the information submitted in accordance with 50.63(c)(1), the Director, Office of Nuclear Reactor Regulation,

will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with § 50.63(a).

(c) Limitation of scope. Paragraph (b) above does not apply to those plants licensed to operate prior to July 21, 1988, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

2. Agency Use of Information

The schedule for implementing equipment modifications is as follows:

(a) For each light-water-cooled nuclear power plant licensed to operate on or before July 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with § 50.63(c)(3), submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of § 50.63(a). This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within 2 years of the notification provided in accordance with § 50.63(c)(3).

(b) A final schedule for implementing modifications necessary to comply with the subject regulations will be established by the NRC staff in consultation and coordination with the affected licensee.

The NRC staff will review the licensees' proposed station blackout duration and the proposed equipment and procedure modifications and their proposed implementation schedule to assure conformance with the rule. The estimated staff time for NRC review of this information is discussed in Item 12 of this Supporting Statement.

3. Reduction in Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this regulation through information technology. Moreover, the NRC encourages the use of such technology.

4. Effort to Identify Duplication

There is no other NRC requirement for licensees to submit information on the length of time that nuclear power plants should be able to withstand a total loss of AC electrical power. Before the resolution of USI A-44 was completed, the NRC requested, as an interim measure (Generic Letter 81-04 dated February 25, 1981), that plants implement, as necessary, emergency procedures and operator training programs to cope with station blackout

events. The information requested by 10 CFR 50.63 is not duplicative of the request for improved procedures and operator training. An issue relating to station blackout was previously cleared under OMB Clearance No. 3150-0067, which expired on May 31, 1983. The purpose of that clearance was to provide specific information in response to a questionnaire on emergency diesel generator operating experience at nuclear power plants and is not duplicative of this information collection requirement.

5. Effort to Use Similar Information

There is no similar information available for the NRC to determine that licensees are capable of effecting emergency procedures and operator training programs to cope with station blackout events.

6. Effort to Reduce Small Business Burden

The provisions of this regulation do not affect small businesses.

7. Consequences of Less Frequent Collection

If the information is not collected as scheduled, the NRC would not be able to assure that a station (nuclear power plant) blackout would not adversely affect the health and safety of the public.

8. Variation from OMB Guidelines

The provisions of this regulation do not vary from OMB Guidelines.

9. Consultations Outside the NRC

None since the consultations that were done during the rulemaking phase.

10. Confidentiality of Information

Information submitted as confidential or proprietary will be handled in accordance with Section 2.790 of the NRC's regulations.

11. Justification for Sensitive Questions

The subject regulations do not require sensitive information.

12. Estimate of Cost to Federal Government

During the period that this clearance is effective, the NRC will only have the Watts Bar Plant to review from the standpoint of the initial and overall review. Based on experience, the staff estimates that approximately 140 hours will be expended within 1 year to complete this review, which will result in a cost of $12,880 (140 hours x $92). In addition, the NRC's consultant will cost approximately $20,000. Annualized over 3 years this is 47 hours x 92 = $4,324 + $6,667 (1/3 of $20,000).

Within the next 3 years, a total of 60 plants will be submitting additional information which was identified in the safety evaluations documenting the initial review. Thus, the NRC estimates that 20 submittals of additional information will be submitted annually. It is also estimated that approximately 80 hours will be expended by the staff for review of each of these submittals. Therefore, the annual burden is estimated to be 1,600 hours, at a cost of $147,200 (1,600 hours x $92).

Based on the above, the annual Federal cost is expected to be:

Expected Submittals	Staff Hours	Cost
Watts Bar (Initial Review)	47	$4,324
Consultant Cost	0	$6,667
Additional Information for 20 Plants	1,600	$147,200
Totals:	1,647	$158,191

13. Estimate of Industry Annual Burden and Cost

Watts Bar is the only plant scheduled to submit the initial and overall station blackout report during the effective period of this clearance. Based on experience, it is estimated that the licensee will expend approximately 1,040 hours within 1 year to prepare and submit the information required by the provisions of 10 CFR 50.63. This would average out to be approximately 347 hours annually. Thus, the annual cost is expected to be $31,924 (347 hours x $92).

Approximately 60 plants will be submitting additional information specified in staff safety evaluation reports during the period this clearance will be effective. This will result in approximately 20 submittals annually. It is estimated that industry will expend approximately 520 hours to prepare and submit the requested information. Thus, annual burden hours are estimated to be 10,400 hours (20 submittals x 520 hours/submittal), at a cost of $956,800 annually (10,400 hours x $92).

Annual industry burden and costs are summarized as follows:

Expected Submittals	Annual Burden Hours	Cost
Watts Bar	347	$31,924
Additional Information (20 plants)	10,400	$956,800
Totals:	10,747	$988,724

14. Reasons for Change in Burden

The estimated burden has been reduced because during the last supporting statement update the NRC estimated 125 responses. During the period this clearance will be effective, the NRC expects approximately 21 responses annually.

15. Publication for Statistical Use

The information collected under 10 CFR 50.63 is not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the subject information collection.

Part 25

SUPPORTING STATEMENT

FOR

DECOMMISSIONING REQUIREMENTS

10 CFR 50.33(k)(1) and (2), 50.75, and 50.82

DESCRIPTION OF THE INFORMATION COLLECTION

The decommissioning regulations specify requirements for financial assurance, recordkeeping and planning and termination procedures. These regulations ensure that decommissioning of production and utilization facilities will be handled by the licensee in a way that will result in minimal or negligible impact on public health and safety and the environment. This regulation affects 122 licensees and construction permit holders for nuclear power plants, (119 operating licenses and construction permit holders for power reactors; Shoreham, Rancho Seco, and Ft. St. Vrain), and 48 licensees for research reactors.

A. JUSTIFICATION

1. Need for the Collection of Information

The provisions of the decommissioning regulations encompass requirements with respect to maintenance of records, submittal and updating as necessary of financial information, either as a certification or plan, and submittal of decommissioning plans.

Section 50.33(k)1 requires that an application for an operating license include information on how reasonable assurance will be provided that funds will be available to decommission the facility.

Section 50.33(k)2 requires holders of operating licenses to provide the above information by July 26, 1990. This information has been supplied.

Section 50.75 establishes detailed information on what the NRC will accept as reasonable assurance that decommissioning funds will be available when needed. This section also specifies requirements for a preliminary decommissioning plan and for recordkeeping.

Section 50.82 requires licensee to submit a decommissioning plan within 2 years after final shutdown of its reactor facility.

2. Agency Use of Information

The NRC will be able to determine if proposed decommissioning can be accomplished without adversely affecting the health and safety of the public. The NRC would also be able to determine whether there would be minimal impact on the environment.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing burden through information technology. Moreover, NRC encourages use of such technology.

4. Effort to Identify Duplication

These requirements are not duplicated in other Federal regulations.

5. Effort to Use Similar Information

Licensees of production and utilization facilities are the only source for this information.

6. Effort to Reduce Small Business Burden

Approximately 3 universities will be required to submit decommissioning plans annually for the next 3 years. There is no way to obtain the necessary information and yet reduce small business burden.

7. Consequences of Less Frequent Collection

There would be a lack of timeliness that could affect proper application of the "as low as is reasonably achievable" principle in carrying out NRC licensing responsibilities.

8. Circumstances Which Justify Variation from OMB Guidelines

The provisions of these regulations do not vary from OMB Guidelines.

9. Consultation Outside the NRC

There has been no consultation with other Federal, State or local government agencies or individuals. However, public comments were solicited during the rulemaking stage.

10. Confidentiality of Information

Confidential submittals are not anticipated. However, confidential or proprietary information would be handled in accordance with 10 CFR 2.790.

11. Justification for Sensitive Questions

The provisions of this regulation do not require sensitive information.

12. Burden for Licensees and the NRC - Decommissioning Reports, Records and Plans

Requirement	Type	Reactor	Licensee hrs. each	Licensee annual burden (responses x hrs)	NRC hrs. each	NRC annual burden (responses x hrs)
50.33(k)(1)	Report	Power	200 hrs.	3 x 200 = 600 hrs.	24 hrs.	3 x 24 = 72 hrs.
50.33(k)(1)	Report	Research	72 hrs.	1 x 72 = 72 hrs.	32 hrs.	1 x 32 = 32 hrs.
50.33(k)(2)*			Complete			
50.75	Record	Power	32 hrs.	122 x 32 = 3904 hrs.	1 hr.	122 x 1 = 122 hrs.
50.75	Record	Research	16 hrs.	48 x 16 = 2000 hrs.	1 hr.	48 x 1 = 48 hrs.
50.82	Report	Power	2000 hrs.	1 x 2000 = 2000 hrs.	800 hrs.	1 x 800 = 800 hrs.
50.82	Report	Research	200 hrs.	3 x 200 = 600 hrs.	100 hrs.	3 x 100 = 300 hrs.

Total Annual Burden: (Licensee) 9,176 hours; (NRC) 1,374 hours
Total Annual Cost: (Licensee) $844,192; (NRC) $126,408 (hours x $92)

*With respect to 50.33(k)(2), all reactor licensees submitted decommissioning funding reports by 7/26/90 as required.

13. Reasons for Change in Burden

The estimated burden for 50.75 and 50.82 represents a decrease because the burden projected for the proposed rule approximately 3 years ago included all plants (110 power; 75 research) that were within the purview of the proposed rule. This supporting statement justifies only the reviews we anticipate (3 research; 1 power reactor) against the final regulation annually for the next 3 years.

14. Publication for Statistical Use

The information collected under the provisions of the subject regulation are not used for statistical purposes.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this collection of information.

Part 26

OMB SUPPORTING STATEMENT

FOR

BANKRUPTCY FILING; NOTIFICATION REQUIREMENTS

10 CFR 50.54(cc)

DESCRIPTION OF THE INFORMATION COLLECTION

Under Section 50.54(cc), promulgated in 1987, licensees are required to notify the appropriate NRC regional office by mail in the event of the commencement of a bankruptcy proceeding involving the licensee. There is no action required of a licensee by these amendments unless and until a bankruptcy petition is filed.

A. JUSTIFICATION

1. Need for the Collection of Information

A licensee who is experiencing severe economic hardship may not be capable of carrying out licensed activities in a manner which protects public health and safety. In particular, a licensee involved in bankruptcy proceedings can have problems affecting payment for proper handling of licensed radioactive material and for decontamination and decommissioning of the licensed facility in a safe manner. Improper materials handling or decontamination activities can lead to spread of contamination throughout a licensee's facility and the potential for dispersion of contaminated material offsite. Financial difficulties can also result in problems affecting the licensee's waste disposal activities.

Instances have occurred in which licensees filed for bankruptcy and the NRC has not been aware that this has happened. NRC inspectors have found belatedly that a licensee has vacated property and abandoned licensed material or that a licensee has been unable to decontaminate its facility and properly dispose of the waste. The NRC is to be notified of these situations promptly so that it can take necessary actions to assure that the health and safety of the public is protected.

2. Agency Use of Information

Notification of the NRC in cases of bankruptcy would alert the NRC so that it may deal with potential hazards to the public health and safety posed by a licensee that does not have the resources to properly secure the licensed material or clean up possible contamination. The information provided by the required notification would be used by the Regional inspection and licensing staff, in consultation with headquarters legal and program staff, to initiate a determination of the need for prompt NRC response or regulatory action. NRC actions may include orders to modify or amend a license or other necessary action and could include limitations on licensed activity which would only permit the storage of licensed

material. The NRC has taken these actions in the past in similar circumstances. In addition, prompt notification of the NRC would allow it to take timely and appropriate action in a bankruptcy proceeding to seek to have available assets of the licensee applied to cover costs of site cleanup before funds are disbursed and become unavailable for cleanup.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection. However, the infrequency of submittal of the information, namely only in those cases where a bankruptcy petition is filed, make use of information technology systems impractical.

4. Effort to Identify Duplication

The Information Requirements Control Automated System (IRCAS) was searched and no duplication was found.

5. Effort to Use Similar Information

There is no similar information available in a form in which it can be used by NRC for the purpose decribed in Item 2. Thus, although a licensee's involvement in a bankruptcy proceeding will be recorded at a bankruptcy court and although the United States Code contains requirements regarding notification of creditors of the commencement of bankruptcy proceedings, this information is not generally available to the NRC in a timely manner so that it can take necessary actions to protect the public health and safety. The resources which would have to be committed by the NRC in monitoring bankruptcy court filings are far in excess of the small burden imposed by this regulation.

6. Effort to Reduce Small Business Burden

All affected licensees are either electric utilities operating power reactors or universities operating research and test reactors. No applications are expected to be received by universities.

7. Consequences of Less Frequent Collection

Information is required to be collected only following the filing of a petition for bankruptcy which is not expected to occur more than one time during the license period of a licensee. If the requested information were not collected at this time, NRC might not be aware of a licensee's significant financial problems. Without this information NRC may not be aware of potential public health and safety problems and not able to act in a timely manner to protect the public health and safety.

8. Circumstances Which Justify Variation from OMB Guidelines

The subject regulation varies from OMB guidelines by requiring that licensees submit the notification in less than 30 days from the date of filing of the petition in bankruptcy. The requirement to provide notification promptly following the filing of the petition is a reasonable measure

to ensure that NRC is made aware of the bankruptcy so as to take effective action to protect the public health and safety. Allowing a period of 30 or more days to elapse might preclude NRC from becoming aware of the licensee's distressed financial circumstances in time to prevent the development or aggravation of a potential hazard to the public. Moreover, the United States Code contains requirements regarding notification of creditors of bankruptcy. This regulation requires one additional notification. Notifying NRC promptly after the filing of the petition would in fact be less of a burden on the bankrupt than a separate notification later in the proceedings.

9. Consultations Outside the NRC

The impetus of this regulation results from cases which have occurred in which licensees have filed for bankruptcy and have vacated property and abandoned licensed material, and NRC inspectors have become aware of this belatedly, resulting in concerns by public and States regarding health and safety. This regulation is intended to address these concerns. Comments were solicited from the public during the proposed rule stage.

10. Confidentiality of Information

Confidential or proprietary information will be protected in accordance with the provisions of 10 CFR and 2.790 and 9.17 of the NRC's regulations.

11. Justification for Sensitive Questions

No sensitive information is requested by this regulation.

12. Estimated Annual Cost to the Federal Government

No burden is expected because no bankruptcy notifications are anticipated at this time.

13. Estimate of Industry Burden

It is estimated that 111 licensees would need approximately 1 hour each to notify the NRC about a bankruptcy filing. However, no industry burden is expected during the effectiveness of the subject OMB clearance because no bankruptcy notifications are anticipated at this time.

14. Reasons for Change in Burden

The current negligible burden estimated for 10 CFR 50.54(cc) is consistent with the negligible burden estimated during OMB review of the proposed rule.

15. Publication for Statistical Use

The information collected under the provisions of the subject regulation are not used for statisical purposes.

B. COLLECTIONS OF INFOMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in this collection of information.

PART 50

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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PART 50 ● DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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[55 FR 29181]

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. 1244, 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. 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 are also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 5035, 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).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); § 50.46 (a) and (b) and 50.54(c) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); § 50.7(a), 50.10(a)–(c), 50.34 (a) and (e), 50.54(a)–(c), 50.46 (a) and (b), 50.47(b), 50.48 (a), (c), (d), and (e), 50.49(a), 50.54 (a), (i), (i)(1), (l)–(n), (p), (q), (t), (v), and (y), 50.55 (f), 50.55 a(a), (c)–(e), (g), and (h), 50.59(c), 50.80(a), 50.82(c), 50.64(b), and 50.80 (a) and (b) are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and § 50.49 (d), (h), and (j), 50.54 (w), (z), (bb), (cc), and (dd), 50.55(e), 50.59(b), 50.61(b), 50.62(b), 50.70(a), 50.71 (a)–(c) and (e), 50.72(a), 50.73 (a) and (b), 50.74, 50.78, and 50.90 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

GENERAL PROVISIONS

§ 50.1 Basis, purpose, and procedures applicable.

[40 FR 8774]

The regulations in this part are promulgated by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended, (68 Stat. 919) and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242) to provide for the licensing of production and utilization facilities.

§ 50.2 Definitions.

[50 FR 34085]

As used in this part,

"Act" means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.

[53 FR 23203]

"Alternate ac source" means an alternating current (ac) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements:

(1) Is connectable to but not normally connected to the offsite or onsite emergency ac power systems;

(2) Has minimum potential for common mode failure with offsite power or the onsite emergency ac power sources;

(3) Is available in a timely manner after the onset of station blackout; and

(4) Has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).

[50 FR 34085]

"Atomic energy" means all forms of energy released in the course of nuclear fission or nuclear transformation.

"Atomic weapon" means any device utilizing atomic energy, exclusive of the means for transporting or propelling the device (where such means is a separable and divisible part of the device), the principal purpose of which is for use as, or for development of, a weapon, a weapon prototype, or a weapon test device.

"By-product material" means any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

"Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.

"Common defense and security" means the common defense and security of the United States.

"Controls" when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor. "Controls" when used with respect to any other facility means apparatus and mechanisms, the manipulation of which could affect the chemical, physical, metallurgical, or nuclear process of the facility in such a manner as to affect the protection of health and safety against radiation.

[53 FR 24018]

"Decommission" means to remove (as a facility) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license.

[50 FR 34085]

"Department" and "Department of Energy" means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565, 42 U.S.C. 7101 *et seq.*), to the extent that the Department, or its duly authorized representatives, exercises functions formerly vested in the Atomic Energy Commission, its Chairman, members, officers and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to sections 104(b), (c) and (d) of the Energy Reorganization Act of 1974 (Pub. L. 93-438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to section 301(a) of the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565 at 577-578, 42 U.S.C. 7151).

[50 FR 34085]

"Design bases" means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

"Electric utility" means any entity that generates or distributes electricity and which recovers the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. Investor-owned utilities, including generation or distribution subsidiaries, public utility districts, municipalities, rural electric cooperatives, and State and Federal agencies, including associations of any of the foregoing, are included within the meaning of "electric utility."

[51 FR 6514]

"Federal Government funding for conversion" means funds appropriated to the Department of Energy or to any other Federal Agency to pay directly to or to reimburse non-power reactor licensees for costs attendant to conversion.

"Fuel acceptable to the Commission" means that the fuel replacing the existing HEU fuel in a specific non-power reactor (1) meets the operating requirements of the existing license or, through appropriate NRC safety review and approval, can be used in a manner which protects public health and safety and promotes the common defense and security; and (2) meets the Commission's policy of limiting, to the maximum extent possible, the use of HEU fuel in that reactor.

[50 FR 34085]

"Government agency" means any executive department, commission, independent establishment, corporation, wholly or partly owned by the United States of America which is an instrumentality of the United States, or any board, bureau, division, service, office, officer, authority, administration, or other establishment in the executive branch of the Government.

"Highly enriched uranium (HEU) fuel" means fuel in which the weight percent of U-235 in the uranium is 20% or greater. Target material, special instrumentation, or experimental devices using HEU are not included.

"Low enriched uranium (LEU) fuel" means fuel in which the weight percent of U-235 in the uranium is less than 20%.

"Non-power reactor" means a research or test reactor licensed under §§ 50.21(c) or 50.22 of this part for research and development.

51 FR 514

"Nuclear reactor" means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction.

"Person" means (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission or the Department, except that the Department shall be considered a person to the extent that its facilities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974,[1] any State or any political subdivision of, or any political entity within a State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (2) any legal successor, representative, agent, or agency of the foregoing.

"Produce," when used in relation to special nuclear material, means (1) to manufacture, make, produce, or refine special nuclear material; (2) to separate special nuclear material from other substances in which such material may be contained; or (3) to make or to produce new special nuclear material.

"Production facility" means:

(1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or

(2) Any facility designed or used for the separation of the isotopes of uranium or the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only; or

50 FR 34085

(3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except (i) laboratory scale facilities designed or used for experimental or analytical purposes, (ii) facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-235 and plutonium produced by the irradiation, if the material processed contains not more than 10^{-4} grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235, and (iii) facilities in which processing is conducted pursuant to a license issued under Parts 30 and 70 of this chapter, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material.

"Reactor coolant pressure boundary" means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment.

(ii) The second of two valves normally closed during normal reactor operation in systen; piping which does not penetrate primary reactor containment,

(iii) The reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping.

"Research and development" means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes, including the experimental production and testing of models, devices, equipment, materials, and processes.

"Restricted Data" means all data concerning (1) design, manufacture, or utilization of atomic weapons; (2) the production of special nuclear material; or (3) the use of special nuclear material in the production of energy, but shall not include data declassified or removed from the Restricted Data category pursuant to section 142 of the Act.

[1] The Department facilities identified in section 202 are:

(1) Demonstration Liquid Metal Fast Breeder reactors when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(2) Other demonstration nuclear reactors, except those in existence on January 19, 1975, when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

"Source material" means source material as defined in subsection 11z. of the Act and in the regulations contained in Part 40 of this chapter.

"Special nuclear material" means (1) plutonium, uranium 233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

[50 FR 34085]

"Station blackout" means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (i.e., single failure) for safe shutdown (non-DBA) is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met. At multi-unit sites, where the combination of emergency ac power sources exceeds the minimum redundancy requirements for safe shutdown (non-DBA) of all units, the remaining emergency ac power sources may be used as alternate ac power sources provided they meet the applicable requirements. If these criteria are not met, station blackout must be assumed on all the units.

[53 FR 23203]

"Testing facility" means a nuclear reactor which is of a type described in § 50.21(c) of this part and for which an application has been filed for a license authorizing operation at:

(1) A thermal power level in excess of 10 megawatts; or

(2) A thermal power level in excess of 1 megawatt, if the reactor is to contain:

(i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or

(ii) A liquid fuel loading; or

(iii) An experimental facility in the core in excess of 16 square inches in cross-section.

[50 FR 34085]

"Unique purpose" means a project, program, or commercial activity which cannot reasonably be accomplished without the use of HEU fuel, and may include: (1) A specific experiment, program, or commercial activity (typically long-term) that significantly serves the U.S. national interest and cannot be accomplished without the use of HEU fuel; (2) Reactor physics or reactor development based explicitly on the use of HEU fuel; (3) Research projects based on neutron flux levels or spectra attainable only with HEU fuel; or (4) A reactor core of special design that could not perform its intended function without using HEU fuel.

[51FR6514]

"Safe shutdown (non-design basis accident (non-DBA))" for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures).

[53 FR 23203]

"United States", when used in a geographical sense, includes Puerto Rico and all territories and possessions of the United States.

"Utilization facility" means any nuclear reactor other than one designed or used primarily for the formation of plutonium or U-233.

[50 FR 34085]

NOTE: Pursuant to subsections 11v. and 11cc., respectively, of the Act, the Commission may from time to time add to, or otherwise alter, the foregoing definitions of production and utilization facility. It may also include as a facility an important component part especially designed for a facility, but has not at this time included any component parts in the definitions.

[51 FR 6514]

§ 50.3 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

[21 FR 355]

§ 50.4 Written communications.

(a) *Address requirements.* The signed original of all correspondence, reports, applications, and other written communications from the applicant or licensee to the Nuclear Regulatory Commission concerning the regulations in this part or individual license conditions must be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555.

(b) *Distribution requirements.* Copies of all correspondence, reports, and other written communications concerning the regulations in this part or individual license conditions must be submitted to the Nuclear Regulatory Commission at the locations and in the quantities set forth below (addresses for the NRC Regional Offices are listed in Appendix D of Part 20 of this chapter).

(1) *Applications for amendment of permits and licenses; reports; and other communications.* All written communications (including responses to generic letters, bulletins, information notices, inspection reports, and miscellaneous requests for additional information), that are required of holders of operating licenses or construction permits issued pursuant to this part, must be submitted as follows, except as otherwise specified in paragraphs (b)(2) through (b)(7) of this section: the signed original to the

[51 FR 40303]

Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility.

(2) *Applications for permits and licenses, and amendments to applications.* Applications for construction permits, applications for operating licenses and amendments to either type of application must be submitted as follows, except as otherwise specified in paragraphs (b)(3) through (b)(7) of this section.

(i) Applications for licenses for facilities described in § 50.21 (a) and (c) and amendments to these applications: The signed original must be sent to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555 and one copy to the appropriate Regional Office.

(ii) Applications for permits and licenses for facilities described in § 50.21(b) or § 50.22, and amendments to these applications: the signed original and 37 copies must be sent to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector, if one has been assigned to the site of the facility.

(3) *Acceptance review application.* Written communications required for an application for determination of suitability for docketing pursuant to § 50.30(a)(8) must be submitted as follows: the signed original and 13 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555 and one copy to the appropriate Regional Office.

(4) *Security plan and related submittals.* Written communications, as defined in paragraphs (b)(4)(i) through (iv) of this section must be submitted as follows: The signed original and three copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, and two copies to the appropriate Regional Office:

(i) Physical security plan pursuant to § 50.34;

(ii) Safeguards contingency plan pursuant to § 50.34;

(iii) Change to security plan, guard training and qualification plan, or safeguards contingency plan made without prior Commission approval pursuant to § 50.54(p);

(iv) Application for amendment of physical security plan, guard training and qualification plan, or safeguards contingency plan pursuant to § 50.90.

(5) *Emergency plan and related submittals.* Written communications as defined in paragraphs (b)(5)(i) through (iii) in this section, must be submitted as follows: the signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, two copies to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility.

(i) Emergency plan pursuant to § 50.34;

(ii) Change to an emergency plan pursuant to § 50.54(q);

(iii) Emergency implementing procedures pursuant to Appendix E.V of this part.

(6) *Updated FSAR.* An updated Final Safety Analysis Report (FSAR) or replacement pages, pursuant to § 50.71(e) must be submitted as follows: the signed original and 10 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility.

(7) *Quality assurance related submittals.* (i) A change to the Safety Analysis Report quality assurance program description pursuant to § 50.54(a)(3) or § 50.55(f)(3), or a change to a licensee's NRC-accepted quality assurance topical report pursuant to § 50.54(a)(3) or § 50.55(f)(3), must be submitted as follows: the signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility.

(ii) A change to an NRC-accepted quality assurance topical report from nonlicensees (i.e., architect/engineers, NSSS suppliers, fuel suppliers, constructors, etc.) must be submitted as follows: one signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555.

(c) *Form of communications.* All copies submitted to meet the requirements set forth in paragraph (b) of this section must be typewritten, printed or otherwise reproduced in permanent form on unglazed paper. Exceptions to these requirements may be granted for the submittal of micrographic, photographic, or electronic forms. Prior to making any submittal in other than paper form, the applicant or licensee must contact the Division of Information Support Services, Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492–8304, to obtain specifications, copy requirements, and prior approval.

(d) *Delivery of communications.* Written communications may be delivered to the Document Control Desk at 11555 Rockville Pike, Rockville, Maryland between the hours of 8:15 a.m. and 4:45 p.m. Eastern Time (ET) (telephone 301–504–1627) on Federal workdays. (Federal workday information can be obtained by calling the Office of Personnel Management at 202–606–2400.)

(e) *Regulation governing submission.* Licensees and applicants submitting correspondence, reports, and other written communications pursuant to the regulations of this part are requested but not required to cite whenever practical, in the upper right corner of the first page of the submittal, the specific regulation or other basis requiring submission.

(f) *Conflicting requirements.* The communications requirements contained in this section and §§ 50.12, 50.30, 50.36, 50.36a, 50.44, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.62, 50.71, 50.73, 50.82, 50.90, and 50.91 supersede and replace all existing requirements in any license conditions or technical specifications in effect on January 5, 1987. Exceptions to these requirements must be approved by the Division of Information Support Services, Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 492–8304.

§ 50.7 Employee protection.

(a) Discrimination by a Commission licensee, permittee, an applicant for a Commission license or permit, or a contractor or subcontractor of a Commission licensee, permittee, or applicant against an employee for engaging in certain protected activities is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, and privileges of employment. The protected activities are established in section 210 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act or the Energy Reorganization Act.

(1) The protected activities include but are not limited to—

(i) Providing the Commission information about possible violations of requirements imposed under either of the above statutes;

(ii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements; or

(iii) Testifying in any Commission proceeding.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.

(3) This section has no application to any employee alleging discrimination

prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in the protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 30 days after an alleged violation occurs by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

(c) A violation of paragraph (a) or paragraph (f) of this section by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant may be grounds for—

(1) Denial, revocation, or suspension of the license.

(2) Imposition of a civil penalty on the licensee or applicant.

(3) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e) Each licensee, permittee and each applicant shall post Form NRC-3, "Notice to Employees," on its premises. Posting must be at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. Premises must be posted not later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, and for 30 days following license termination.

Note.—Copies of Form NRC-3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in Appendix D, Part 20 of this chapter.

(f) No agreement affecting the compensation, terms, conditions and privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 210 of the Energy Reorganization Act of 1974, may contain any provision which would prohibit, restrict, or otherwise discourage, an employee from participating in protected activity as defined in paragraph (a)(1) of this section, including, but not limited to, providing information to the NRC on potential violations or other matters within NRC's regulatory responsibilities.

§ 50.8 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). OMB has approved the information collection requirements contained in this part under control number 3150-0011.

(b) The approved information collection requirements contained in this part appear in §§ 50.30, 50.33, 50.33a, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.46, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.63, 50.64, 50.71, 50.72, 50.80, 50.82, 50.90, 50.91, and Appendixes A, B, E, G, H, J, K, M, N, and R.

(c) This part contains information collection requirements in addition to those approved under the control number specified in paragraph (a) of this section. These information collection requirements and the control numbers under which they are approved are as follows:

(1) In § 50.73, Form NRC-366 is approved under control number 3150-0104.

(2) In § 50.78, Form N-71 is approved under control number 3150-0056.

§ 50.9 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

(b) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

REQUIREMENT OF LICENSE, EXCEPTIONS

§ 50.10 License required.

(a) Except as provided in § 50.11, no person within the United States shall transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use any production or utilization facility except as authorized by a license issued by the Commission.

(b) No person shall begin the construction of a production or utilization facility on a site on which the facility is to be operated until a construction permit has been issued. As used in this paragraph, the term "construction" shall be deemed to include pouring the foundation for, or the installation of, any portion of the permanent facility on the site, but does not include:

(1) Site exploration, site excavation, preparation of the site for construction of the facility, including the driving of piles, and construction of roadways, railroad spurs, and transmission lines;

(2) Procurement or manufacture of components of the facility;

(3) Construction of non-nuclear facilities (such as turbogenerators and turbine buildings) and temporary buildings (such as construction equipment storage sheds) for use in connection with the construction of the facility; and

(4) With respect to production or utilization facilities, other than testing facilities, required to be licensed pursuant to section 104a or section 104c of the Act, the construction of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility. (For example, the construction of a college laboratory building with space for installation of a training reactor is not affected by this paragraph.

This paragraph does not apply to production or utilization facilities subject to paragraph (c) of this section.

(c) Notwithstanding the provisions of paragraph (b) of this section, and subject to paragraphs (d) and (e) of this section, no person shall effect commencement of construction of a production or utilization facility subject to the provisions of § 51.20(b) of this chapter on a site on which the facility is to be operated until a construction permit has been issued. As used in this paragraph, the term "commencement of construction" means any clearing of land, excavation or other substantial action that would adversely affect the environment of a site, but does not mean:

55 FR 10397 — 51 FR 40303 — 47 FR 30452 — 55 FR 10397 — 49 FR 19623 — 53 FR 23203 — 49 FR 19623 — 52 FR 49362 — 43 FR 6915 — 33 FR 2381 — 25 FR 8712 — 37 FR 5745 — 49 FR 9352

(1) Changes desirable for the temporary use of the land for public recreational uses, necessary borings to determine foundation conditions or other preconstruction monitoring to establish background information related to the suitability of the site or to the protection of environmental values;

(2) Procurement or manufacture of components of the facility; and

(3) With respect to production or utilization facilities, other than testing facilities, required to be licensed pursuant to section 104a or section 104c of the Act, the construction of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility. (For example, the construction of a college laboratory building with space for installation of a training reactor is not affected by this paragraph.)

(d)(1) Each person subject to the provisions of paragraph (c) of this section, who is, on March 21, 1972, conducting activities permitted pursuant to paragraph (b) of this section in effect prior to March 21, 1972, may furnish to the Commission within 30 days after March 21, 1972 or such later date as may be approved by the Commission upon good cause shown, a written statement of any reasons, with supporting factual submission, why, with reference to the factors stated in paragraph (d)(2) of this section, the activities should be continued, pending the issuance of a construction permit, notwithstanding the provisions of paragraph (c) of this section. If such written statement has been submitted within the time specified, such activities may continue to be conducted pending Commission action pursuant to paragraph (d)(2) of this section.

(2) Upon submission of a statement of reasons pursuant to paragraph (d)(1) of this section the Commission may authorize the continued conduct of activities permitted by paragraph (b) of this section in effect prior to March 21, 1972, upon consideration and balancing of the following factors:

(i) Whether continuation of the activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;

(ii) Whether redress of any adverse environmental impact from continuation of the activities can reasonably be effected should such redress be necessary;

(iii) Whether continuation of the activities would foreclose subsequent adoption of alternatives; and

(iv) The effect of delay in conducting such activities on the public interest, including the power needs to be served by the proposed facility, the availability of alternative sources, if any, to meet those needs on a timely basis, and delay costs to the applicant and to consumers.

(3) Activities permitted to be continued pursuant to this paragraph (d) shall be conducted in such a manner as will minimize or reduce their environmental impact.

(e)(1) The Director of Nuclear Reactor Regulation may authorize an applicant for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in §§ 50.21(b) (2) or (3) or 50.22 or is a testing facility to conduct the following activities: (i) Preparation of the site for construction of the facility (including such activities as clearing, grading, construction of temporary access roads and borrow areas); (ii) installation of temporary construction support facilities (including such items as warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and construction support buildings); (iii) excavation for facility structures; (iv) construction of service facilities (including such facilities as roadways, paving, railroad spurs, fencing, exterior utility and lighting systems, transmission lines, and sanitary sewerage treatment facilities); and (v) the construction of structures, systems and components which do not prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. No such authorization shall be granted unless the staff has completed a final environmental impact statement or the issuance of the construction permit as required by Subpart A of Part 51 of this chapter.

(2) Such an authorization shall be granted only after the presiding officer in the proceeding on the construction permit application (i) has made all the findings required by §§ 51.104(b) and 51.105 of this chapter to be made prior to issuance of the construction permit for the facility, and (ii) has determined that, based upon the available information and review to date, there is reasonable assurance that the proposed site is a suitable location for a reactor of the general size and type proposed from the standpoint of radiological health and safety considerations under the Act and rules and regulations promulgated by the Commission pursuant thereto.

(3)(i) The Director of Nuclear Reactor Regulation may authorize an applicant for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter, and is of the type specified in §§ 50.21(b) (2) or (3) or 50.22 or is a testing facility to conduct, in addition to the activities described in paragraph (e)(1) of this section, the installation of structural foundations, including any necessary subsurface preparation, for structures, systems and components which prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

(ii) Such an authorization, which may be combined with the authorization described in paragraph (e)(1) of this section, or may be granted at a later time, shall be granted only after the presiding officer in the proceeding on the construction permit application has, in addition to making the findings and determinations required by paragraph (e)(2) of this section, determined that there are no unresolved safety issues relating to the additional activities that may be authorized pursuant to this paragraph that would constitute good cause for withholding authorization.

(4) Any activities undertaken pursuant to an authorization granted under this paragraph shall be entirely at the risk of the applicant and, except as to matters determined under paragraphs (e)(2) and (e)(3)(ii), the grant of the authorization shall have no bearing on the issuance of a construction permit with respect to the requirements of the Act, and rules, regulations, or orders promulgated pursuant thereto.

§ 50.11 Exceptions and exemptions from licensing requirements.

Nothing in this part shall be deemed to require a license for:

(a) The manufacture, production, or acquisition by the Department of Defense of any utilization facility authorized pursuant to section 91 of the Act, or the use of such facility by the Department of Defense or by a person under contract with and for the account of the Department of Defense;

(b) Except to the extent that Administration facilities of the types subject to licensing pursuant to section 202 of the Energy Reorganization Act of 1974[1] are involved;

[1]The Department facilities identified in section 202 are:

(1) Demonstration Liquid Metal Fast Breeder reactors when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(2) Other demonstration nuclear reactors, except those in existence on January 19, 1975, when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(1)(i) The processing, fabrication or refining of special nuclear material or the separation of special nuclear material, or the separation of special nuclear material from other substances by a prime contractor of the Department under a prime contract for:

(A) The performance of work for the Department at a United States government-owned or controlled site;

(B) Research in, or development, manufacture, storage, testing or transportation of, atomic weapons or components thereof; or

(C) The use or operation of a production or utilization facility in a United States owned vehicle or vessel; or

(ii) By a prime contractor or subcontractor of the Commission or the Department under a prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law, and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety;

(2)(i) The construction or operation of a production or utilization facility for the Department at a United States government-owned or controlled site, including the transportation of the production or utilization facility to or from such site and the performance of contract services during temporary interruptions of such transportation; or the construction or operation of a production or utilization facility for the Department in the performance of research in, or development, manufacture, storage, testing, or transportation of, atomic weapons or components thereof; or the use or operation of a production or utilization facility for the Department in a United States government-owned vehicle or vessel: *Provided*, That such activities are conducted by a prime contractor of the Department under a prime contract with the Department.

(ii) The contruction or operation of a production or utilization facility by a prime contractor or subcontractor of the Commission or the Department under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontrator is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety.

(c) The transportation or possession of any production or utilization facility by a common or contract carrier or warehousemen in the regular course of carriage for another or storage incident thereto.

[40 FR 8774]

§ 50.12 Specific exemptions.

(a) The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part, which are—

(1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

(2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever—

(i) Application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission; or

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated; or

[50 FR 50764]

(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption; or

(v) The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation; or

(vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. If such condition is relied on exclusively for satisfying paragraph (a)(2) of this section, the exemption may not be granted until the Executive Director for Operations has consulted with the Commission.

(b) Any person may request an exemption permitting the conduct of activities prior to the issuance of a construction permit prohibited by § 50.10. The Commission may grant such an exemption upon considering and balancing the following factors:

(1) Whether conduct of the proposed activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;

(2) Whether redress of any adverse environment impact from conduct of the proposed activities can reasonably be effected should such redress be necessary;

(3) Whether conduct of the proposed activities would foreclose subsequent adoption of alternatives; and

(4) The effect of delay in conducting such activities on the public interest, including the power needs to be used by the proposed facility, the availability of alternative sources, if any, to meet those needs on a timely basis and delay costs to the applicant and to consumers.

[50 FR 50764]

Issuance of such an exemption shall not be deemed to constitute a commitment to issue a construction permit. During the period of any exemption granted pursuant to this paragraph (b), any activities conducted shall be carried out in such a manner as will minimize or reduce their environmental impact.

§ 50.13 Attacks and destructive acts by enemies of the United States; and defense activities.

[32 FR 13345]

An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is not required to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person, or (b) use or deployment of weapons incident to U.S. defense activities.

CLASSIFICATION AND DESCRIPTION OF LICENSES

[21 FR 355]

§ 50.20 Two classes of licenses.

Licenses will be issued to named persons applying to the Commission therefor, and will be either class 104 or class 103.

§ 50.21 Class 104 licenses; for medical therapy and research and development facilities.

[43 FR 6915]

A class 104 license will be issued, to an applicant who qualifies, for any one or more of the following: to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use,

(a) A utilization facility for use in medical therapy; or

[35 FR 19656]

(b)(1) A production or utilization facility the construction or operation of which was licensed pursuant to subsection 104b of the Act prior to December 19, 1970;

(2) A production or utilization facility for industrial or commercial purposes constructed or operated under an arrangement with the Administration entered into under the Cooperative Power Reactor Demonstration Program, except as otherwise specifically required by applicable law; and

(3) A production or utilization facility for industrial or commercial purposes, when specifically authorized by law.

[48 FR 6915]

(c) A production or utilization facility, which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, and which is not a facility of the type specified in paragraph (b) of this section or in § 50.22.

[48 FR 6915]

§ 50.22 Class 103 licenses; for commercial and industrial facilities.

A class 103 license will be issued, to an applicant who qualifies, for any one or more of the following: To transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use a production or utilization facility for industrial or commercial purposes; *Provided, however,* That in the case of a production or utilization facility which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training.

[48 FR 6915]

§ 50.23 Construction permits.

A construction permit for the construction of a production or utilization facility will be issued prior to the issuance of a license if the application is otherwise acceptable, and will be converted upon due completion of the facility and Commission action into a license as provided in § 50.56 of this part. A construction permit for the alteration of a production or utilization facility will be issued prior to the issuance of an amendment of a license, if the application for amendment is otherwise acceptable, as provided in § 50.91.

§ 50.24 [Deleted]

APPLICATIONS FOR LICENSES, FORM, CONTENTS, INELIGIBILITY OF CERTAIN APPLICANTS

[51 FR 40307]

§ 50.30 Filing of application for licenses; oath or affirmation.

(a) *Serving of applications.* (1) Each filing of an application for a license to construct and/or operate a production or utilization facility (including amendments to the applications) must be submitted to the U.S. Nuclear Regulatory Commission in accordance with § 50.4.

(2) An additional 10 copies of the general information and 30 copies of the safety analysis report, or part thereof or amendment thereto, must be retained by the applicant for distribution in accordance with the written instructions of the Director, Office of Nuclear Reactor Regulation, or the Director, Office of Nuclear Material Safety and Safeguards, as appropriate.

[51 FR 40307]

(3) Each applicant shall, upon notification by the Atomic Safety and Licensing Board appointed to conduct the public hearing required by the Atomic Energy Act for the issuance of a construction permit, update the application and serve the updated copies of the application or parts of it, eliminating all superseded information, together with an index of the updated application, as directed by the Atomic Safety and Licensing Board. In addition, at that time the applicant shall serve a copy of the updated application on the Atomic Safety and Licensing Appeal Panel. Any subsequent amendment to the application must be served on those served copies of the application and must be submitted to the U.S. Nuclear Regulatory Commission as specified in § 50.4.

[51 FR 40307]

(4) The applicant must make a copy of the updated application available at the public hearing for the use of any other parties to the proceeding, and shall certify that the updated copies of the application contain the current contents of the application submitted in accordance with the requirements of this part.

(5) At the time of filing an application, the Commission will establish a Local Public Document Room near the site of the proposed facility, for the use of the public, where a copy of the application, subsequent amendments, and other records pertinent to the facility will be available for public inspection and copying.

(6) The serving of copies required by this section must not occur until the application has been docketed pursuant to § 2.101(a) of this chapter. Copies must be submitted to the Commission, as specified in § 50.4, to enable the Director, Office of Nuclear Reactor Regulation, or the Director, Office of Nuclear Material Safety and Safeguards, as appropriate, to determine whether the application is sufficiently complete to permit docketing.

(b) *Oath or affirmation.* Each application for a license, including whenever appropriate a construction permit, or amendment of it, and each amendment of each application must be executed in a signed original by the applicant or duly authorized officer thereof under oath or affirmation.

(c) [Removed] [51 FR 40307]

(d) *Application for operating licenses.* The holder of a construction permit for a production or utilization facility shall, at the time of submission of the final safety analysis report, file an application for an operating license or an amendment to an application for a license to construct and operate a production or utilization facility for the issuance of an operating license, as appropriate. The application or amendment shall state the name of the applicant, the name, location and power level, if any, of the facility and the time when the facility is expected to be ready for operation, and may incorporate by reference any pertinent information submitted in accordance with § 50.33 with the application for a construction permit.

(e) *Filing fees.* Each application for a production or utilization facility license, including, whenever appropriate, a construction permit, other than a license exempted from Part 170 of this chapter, shall be accompanied by the fee prescribed in Part 170 of this chapter. No fee will be required to accompany an application for renewal, amendment or termination of a construction permit or operating license, except as provided in § 170.21 of this chapter.

(f) *Environmental report.* An application for a construction permit or an operating license for a nuclear power reactor, testing facility, fuel reprocessing plant, or such other production or utilization facility whose construction or operation may be determined by the Commission to have a significant impact on the environment shall be accompanied by any Environmental Report required pursuant to Subpart A of Part 51 of this chapter.

§ 50.31 **Combining applications.**

An applicant may combine in one his several applications for different kinds of licenses under the regulations in this chapter.

§ 50.32 **Elimination of repetition.**

In his application, the applicant may incorporate by reference information contained in previous applications, statements or reports filed with the Commission: *Provided*, That such references are clear and specific.

§ 50.33 **Contents of applications; general information.**

Each application shall state:

(a) Name of applicant;

(b) Address of applicant;

(c) Description of business or occupation of applicant;

(d) (1) If applicant is an individual, state citizenship.

(2) If applicant is a partnership, state name, citizenship and address of each partner and the principal location where the partnership does business.

(3) If applicant is a corporation or an unincorporated association, state:

(i) The state where it is incorporated or organized and the principal location where it does business.

(ii) The names, addresses and citizenship of its directors and of its principal officers.

(iii) Whether it is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government, and if so, give details.

(4) If the applicant is acting as agent or representative of another person in filing the application, identify the principal and furnish information required under this paragraph with respect to such principal.

(e) The class of license applied for, the use to which the facility will be put, the period of time for which the license is sought, and a list of other licenses, except operator's licenses, issued or applied for in connection with the proposed facility.

(f) Except for an electric utility applicant for a license to operate a utilization facility of the type described in § 50.21(b) or § 50.22, information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations in this chapter, the activities for which the permit or license is sought. As applicable, the following should be provided:

(1) If the application is for a construction permit, the applicant shall submit information that demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs. The applicant shall submit estimates of the total construction costs of the facility and related fuel cycle costs, and shall indicate the source(s) of funds to cover these costs.

(2) If the application is for an operating license, the applicant shall submit information that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license. The applicant shall submit estimates for total annual operating costs for each of the first five years of operation of the facility. The applicant shall also indicate the source(s) of funds to cover these costs. An application to renew or extend the term of an operating license must include the same financial information as is required in an application for an initial license.

(3) Each application for a construction permit or an operating license submitted by a newly-formed entity organized for the primary purpose of constructing or operating a facility must also include information showing:

(i) The legal and financial relationships it has or proposes to have with its stockholders or owners;

(ii) Its financial ability to meet any contractual obligation to the entity which they have incurred or proposed to incur; and

(iii) Any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification.

(4) The Commission may request an established entity or newly-formed entity to submit additional or more detailed information respecting its financial arrangements and status of funds if the Commission considers this information appropriate. This may include information regarding a licensee's ability to continue the conduct of the activities authorized by the license and to decommission the facility.

45 FR 55402

(g) If the application is for an operating license for a nuclear power reactor, the applicant shall submit radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within the plume exposure pathway Emergency Planning Zone (EPZ),[1] as well as the plans of State governments wholly or partially within the ingestion pathway EPZ.[2] Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to the local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

21 FR 355

(h) If the applicant proposes to construct or alter a production or utilization facility, the application shall state the earliest and latest dates for completion of the construction or alteration.

[1] Emergency Planning Zones (EPZs) are discussed in NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," December 1978.

[2] If the State and local emergency response plans have been previously provided to the NRC for inclusion in the facility docket, the applicant need only provide the appropriate reference to meet this requirement.

36 FR 19555

(i) If the proposed activity is the generation and distribution of electric energy under a class 103 license, a list of the names and addresses of such regulatory agencies as may have jurisdiction over the rates and services incident to the proposed activity, and a list of trade and news publications which circulate in the area where the proposed activity will be conducted and which are considered appropriate to give reasonable notice of the application to those municipalities, private utilities, public bodies, and cooperatives, which might have a potential interest in the facility.

21 FR 355

(j) If the application contains Restricted Data or other defense information, it shall be prepared in such manner that all Restricted Data and other defense information are separated from the unclassified information.

53 FR 24018

(k)(1) For an application for an operating license for a production or utilization facility, information in the form of a report, as described in § 50.75 of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

(2) On or before July 26, 1990, each holder of an operating license for a production or utilization facility in effect on July 27, 1990, shall submit information in the form of a report as described in § 50.75 of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

§ 50.33a Information requested by the Attorney General for antitrust review.

44 FR 60715

(a)(1) An applicant for a construction permit for a nuclear power reactor shall submit the information requested by the Attorney General as described in Appendix L to this part, if the application is for a class 103 permit and if the applicant has electrical generating capacity exceeding 1400 MW(e).

(2) An applicant for a construction permit for a nuclear power reactor shall submit the information requested by the Attorney General as described in paragraph 9 of Section II of Appendix L if the applicant has electrical generating capacity exceeding 200 MW(e) but no more than 1400 MW(e). Upon request of the Commission, the applicant shall furnish the other information described in Appendix L.

(3) An applicant for a construction permit for a nuclear power reactor is not required to submit the information described in Appendix L unless specifically requested by the Commission to provide the information, if the applicant has electrical generating capacity of 200 MW(e) or less.

44 FR 60715

(4) The information described in paragraphs (a)(1) and (2) of this section shall be submitted as a separate document prior to any other part of the license application as provided in paragraph (b) and in accordance with § 2.101 of this chapter.

42 FR 22882

(b) Except as provided in paragraph (d), any person who applies for a class 103 construction permit for a nuclear power reactor on or after July 28, 1975 shall submit the document titled "Information Requested by the Attorney General for Antitrust Review" at least nine (9) months but not more than thirty-six months prior to the date of submittal of any part of the application for a class 103 construction permit.

(c) Revoked 43 FR 49775

42 FR 22882

(d) Any person who applies for a class 103 construction permit for a nuclear power reactor pursuant to the provisions of § 2.101(a-1) and Subpart F of Part 2 of this chapter shall submit the document title "Information Requested by the Attorney General for Antitrust Review" at least nine (9) months but not more than thirty-six months prior to the filing of part two or part three of the application, whichever part is filed first, as specified in § 2.101(a-1) of this chapter.

44 FR 60715

(e) Any person who applies for a class 103 construction permit for a uranium enrichment or fuel reprocessing plant shall submit such information as may be requested by the Attorney General for antitrust review, as a separate document as soon as possible and in accordance with § 2.101 of this chapter.

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report.* Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[4] to be included shall consist of the following:

(1) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in Part 100 of this chapter. Such assessment shall contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in Part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit.

(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[4] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units.

(ii) The design bases and the relation of the design bases to the principal design criteria.

(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (ii) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 of this part for facilities for which construction permits may be issued after December 28, 1974.

(5) An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: *Provided, however,* That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.

(6) A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.

(7) A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.

(8) An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

(9) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(10) A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.

(11) On or after February 5, 1979, applicants who apply for construction permits for nuclear powerplants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls that will be used during construction to assure the safety of the operating unit.

(b) *Final safety analysis report.* Each application for a license to operate a facility shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:

(1) All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the construction permit, relating to site evaluation factors identified in Part 100 of this chapter.

(2) A description and analysis of the structures, systems, and components

[4] The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.

[4] General design criteria for chemical processing facilities are being developed.

of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report.

Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974.

(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.

(iii) Plans for preoperational testing and initial operations.

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

(v) Plans for coping with emergencies, which shall include the items specified in Appendix E.

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of Part 55 of this chapter.

(9) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61 (b)(1) and (b)(2).

(c) *Physical security plan.* Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 (and Part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a), at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable.

(d) *Safeguards contingency plan.* Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50, 73.55, or 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in Appendix C to 10 CFR Part 73 are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)[1]

(e) Each applicant for a license to operate a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate.

[1] A physical security plan that contains all the information required in both § 73.55 and Appendix C to Part 73 satisfies the requirement for a contingency plan.

of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. [33 FR 18610]

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report.

Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974. [39 FR 1002]

(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved. [33 FR 18610]

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied. [35 FR 18301]

(iii) Plans for preoperational testing and initial operations. [41 FR 6256]

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components. [41 FR 19661]

(v) Plans for coping with emergencies, which shall include the items specified in Appendix E. [35 FR 19556]

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36. [33 FR 18610]

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites. [43 FR 34764]

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter. [36 FR 6538]

(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of Part 55 of this chapter. [54 FR 9453]

(9) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61 (b)(1) and (b)(2). [60 FR 65456]

(c) *Physical security plan.* Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 (and Part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a), at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable. [45 FR 75948]

(d) *Safeguards contingency plan.* Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50, 73.55, or 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in Appendix C to 10 CFR Part 73 are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)[1] [51 FR 27817]

(e) Each applicant for a license to operate a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate. [46 FR 52716]

[1] A physical security plan that contains all the information required in both § 73.55 and Appendix C to Part 73 satisfies the requirement for a contingency plan.

during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

(xiv) Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system.

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series.

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal.

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation.

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to

of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. [33 FR 18610]

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report.

Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974. [39 FR 1002]

(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved. [33 FR 18610]

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied. [36 FR 18384]

(iii) Plans for preoperational testing and initial operations. [41 FR 56561]

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components. [41 FR 56561]

(v) Plans for coping with emergencies, which shall include the items specified in Appendix E. [35 FR 19556]

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36. [33 FR 18610]

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites. [43 FR 24764]

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter. [36 FR 16536]

(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of Part 55 of this chapter. [52 FR 9453]

(9) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61 (b)(1) and (b)(2). [52 FR 47793]

(c) *Physical security plan.* Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 (and Part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a), at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable. [45 FR 26948]

(d) *Safeguards contingency plan.* Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50, 73.55, or 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in Appendix C to 10 CFR Part 73 are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)[1] [51 FR 27817]

(e) Each applicant for a license to operate a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate. [46 FR 51718]

[1] A physical security plan that contains all the information required in both § 73.55 and Appendix C to Part 73 satisfies the requirement for a contingency plan.

(f) *Additional TMI-related requirements.* In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982 shall meet the requirements in paragraphs (f)(1) through (3) of this section. This rule applies only to the pending applications by Duke Power Company (Perkins Nuclear Station Units 1, 2 and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1 and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants). The number of units that will be specified in the manufacturing license, if issued, will be that number whose start of manufacture as defined in the license application, can practically begin within a ten-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than five years after its approval. Paragraphs (f)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this rule. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and Criterion 50 of Appendix A to 10 CFR Part 50, is sufficient for issuance of the manufacturing license may be considered in the manufacturing license proceeding.

(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license.[1]

(i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.E.1.1):

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

(C) An evaluation of AFWS flow design bases and criteria.

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)

(v) Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

(vii) Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)

(xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only) (I.A.4.2)

(ii) Establish a program, to begin

[1] Alphanumeric designations correspond to the related action plan items in NUREG 0718 and NUREG 0660 "NRC Action Plan Developed as a Result of the TMI-2 Accident." They are provided herein for information only.

during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g. noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolent system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

(xiv) Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system.

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series.

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal.

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation.

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only) (II.G.1)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to

ensure proper functioning when the ....in feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)

(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)

(xxiii) Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)

(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)

(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

(xxvii) Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. (III.D.3.4)

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.

(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)

(ii) Ensure that the ...uality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety. (I.F.1)

(iii) Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)

(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A)(*1*) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(*2*) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(*1*) and (f)(3)(v)(B)(*1*) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register**. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L St., NW., Washington, D.C.

(B) (*1*) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category. (*2*) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D)

action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under § 50.21(b) or § 50.22, the licensee shall notify the Commission if required by § 50.72 and shall submit a Licensee Event Report to the Commission as required by § 50.73. In this case licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications.

(3) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met.

(4) *Design features.* Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

(5) *Administrative controls.* Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

(6) *Initial Notification.* Reports made to the Commission by licensees in response to the requirements of this section must be made as follows:

(i) Licensees that have an installed Emergency Notification System shall make the initial notification to the NRC Operations Center in accordance with § 50.72 of this part.

(ii) All other licensees shall make the initial notification by telephone to the Administrator of the appropriate NRC Regional Office listed in Appendix D, Part 20, of this chapter.

(7) *Written Reports.* Licensees for nuclear power reactors licensed under § 50.21(b) and § 50.22 of this part shall submit written reports to the Commission in accordance with § 50.73 of this part for events described in paragraphs (c)(1) and (c)(2) of this section. For all licensees, the Commission may require Special Reports as appropriate.

(d)(1) This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.

(2) An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969.

(3) At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.

§ 50.36a Technical specifications on effluents from nuclear power reactors.

(a) In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.106 of this chapter, require:

(1) That operating procedures developed pursuant to § 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system, pursuant to § 50.34a(a), be maintained and used. The licensee shall retain the operating procedures in effect as a record until the Commission terminates the reactor license and shall retain each superseded revision of the procedures for three years from the date it was superseded.

(2) Each licensee shall submit a report to the Commission within 60 days after January 1 and July 1 of each year that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous six months of operation, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. The report must be submitted as specified in § 50.4. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report must cover this specifically. On the basis of these reports and any additional information the Commission may obtain from the licensee or others, the Commission may require the licensee to take action as the Commission deems appropriate.

(b) In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in § 20.106 of this chapter and in the operating license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in § 20.106 of this chapter and the operating license. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable.

The guides set out in Appendix I provide numerical guidance on limiting conditions for operation for light-water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

§ 50.36b Environmental conditions.

Each license authorizing operation of a production or utilization facility which is of a type described in § 50.21(b) (2) or (3) or § 50.22 or is a testing facility may include conditions to protect the environment to be set out in an attachment to the license which is incorporated in and made a part of the license. These conditions will be derived from information contained in the environmental report and the supplement to the environmental report submitted pursuant to §§ 51.50 and 51.53 of this chapter as analyzed and evaluated in the NRC record of decision, and will identify the obligations of the licensee in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirements for the protection of the nonaquatic environment.

[21 F.R. 355]

§ 50.37 Agreement limiting access to Restricted Data.

As part of his application and in any event prior to the receipt of Restricted Data or the issuance of a license or construction permit, the applicant shall agree in writing that he will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Commission on the character, associations, and loyalty of such individual, and the Commission shall have determined that permitting such person to have access to Restricted Data will not endanger the common defense and security. The agreement of the applicant in this regard shall be deemed part of the license or construction permit, whether so stated therein or not.

[21 F.R. 6815]

§ 50.38 Ineligibility of certain applicants.

Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license.

[21 F.R. 355]

§ 50.39 Public inspection of applications.

Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with the provisions of the regulations contained in Part 2 of this chapter.

STANDARDS FOR LICENSES AND CONSTRUCTION PERMITS

§ 50.40 Common standards.

In determining that a license will be issued to an applicant, the Commission will be guided by the following considerations:

(a) The processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20, and that the health and safety of the public will not be endangered.

(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.

(c) Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete final safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in this chapter.

§ 50.36 Technical specifications.

(a) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(b) Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

(c) Technical specifications will include items in the following categories:

(1) *Safety limits, limiting safety system settings, and limiting control settings.* (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and that must not be exceeded in order to protect the integrity of the physical system that is designed to guard against the uncontrolled release of radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action must be taken as stated in the technical specification or the affected part of the process, or the entire process if required, must be shut down, unless this action would further reduce the margin of safety. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. If a portion of the process or the entire process has been shutdown, operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(2) *Limiting conditions for operation.* Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under § 50.21(b) or § 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective

proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

(9) Conformance with the Standard Review Plan (SRP).

(1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(ii) Applications for light water cooled nuclear power plant construction permits, manufacturing licenses, and preliminary or final design approvals for standard plants docketed after May 17, 1982 shall include an evaluation of the facility against the SRP in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

(3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

§ 50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.

(a) An application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest. The guides set out in Appendix I to this part provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirements that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.

(b) Each application for a permit to construct a nuclear power reactor shall include:

(1) A description of the preliminary design of equipment to be installed pursuant to paragraph (a) of this section;

(2) An estimate of:

(i) The quantity of each of the principal radio-nuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and

(ii) The quantity of each of the principal radio-nuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal reactor operations.

(3) A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

(c) Each application for a license to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, pursuant to paragraph (a) of this section; an (2) a revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.

§ 50.35 Issuance of construction permits.[1]

(a) When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public; (2) such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report; (3) safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in Part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.

[1] The Commission may issue a provisional construction permit pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional construction permit has been published on or before that date.

(b) The applicant is technically and financially qualified to engage in the proposed activities in accordance with the regulations in this chapter. However, no consideration of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in § 50.21(b) or § 50.22. [49 FR 35747]

(c) The issuance of a license to the applicant will not, in the opinion of the Commission, be inimical to the common defense and security or to the health and safety of the public. [21 FR 355]

(d) Any applicable requirements of subpart A of Part 51 have been satisfied. [40 FR 8352]

§ 50.41 Additional standards for class 104 licenses.

In determining that a class 104 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in § 50.40 be guided by the following considerations: [21 FR 355]

(a) The Commission will permit the widest amount of effective medical therapy possible with the amount of special nuclear material available for such purposes.

(b) The Commission will permit the conduct of widespread and diverse research and development.

(c) An application for a class 104 operating license as to which a person who intervened or sought by timely written notice to the Commission to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination has requested an antitrust review under section 105 of the Act within 25 days after the date of publication in the FEDERAL REGISTER of notice of filing of the application for an operating license or December 19, 1970, whichever is later, is also subject to the provisions of § 50.42(b). [35 FR 19556]

§ 50.42 Additional standards for class 103 licenses.

In determining whether a class 103 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in § 50.40, be guided by the following considerations: [21 FR 355]

(a) The proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized.

(b) Due account will be taken of the advice provided by the Attorney General, pursuant to subsection 105c of the Act, and to such evidence as may be provided during any proceedings in connection with the antitrust aspects of the application. For this purpose, the Commission will promptly transmit to the Attorney General a copy of the license application, and request such advice as the Attorney General determines to be appropriate in regard to the finding to be made by the Commission as to whether the proposed license would create or maintain a situation inconsistent with the antitrust laws, as specified in subsection 105a of the Act: *Provided*, That this requirement will not apply with respect to the types of class 103 licenses which the Commission, with the approval of the Attorney General, may determine would not significantly affect the applicant's activities under the antitrust laws: *And provided further*, That this requirement will not apply to an application for a license to operate a production or utilization facility for which a class 103 construction permit was issued unless the Commission, after consultation with the Attorney General, determines such review is advisable on the ground that significant changes in the licensee's activities or proposed activities have occurred subsequent to the previous review by the Attorney General and the Commission. Upon receipt of the Attorney General's advice, the Commission will cause such advice to be published in the FEDERAL REGISTER. After consideration of the antitrust aspects of the application, the Commission, if it finds that the license to be issued or continued, would create or maintain a situation inconsistent with the antitrust laws as specified in subsection 105a of the Act, will consider, in determining whether a license should be issued or continued, such other factors as the Commission in its judgment deems necessary to protect the public interest, including the need for power in the affected area.[1] [35 FR 19556] [21 FR 355]

§ 50.43 Additional standards and provisions affecting class 103 licenses for commercial power.

In addition to applying the standards set forth in §§ 50.40 and 50.42, in the case of a class 103 license for a facility for the generation of commercial power:

(a) The Commission will give notice in writing of each application to such regulatory agency as may have jurisdiction over the rates and services incident to the proposed activity; will publish notice of the application in such trade or news publications as it deems appropriate to give reasonable notice to municipalities, private utilities, public bodies, and cooperatives which might have a potential interest in such utilization or production facility; and will publish notice of the application once each week for 4 consecutive weeks in the FEDERAL REGISTER. No license will be issued by the Commission prior to the giving of such notices and until 4 weeks after the last publication in the FEDERAL REGISTER. [35 FR 19556]

(b) If there are conflicting applications for a limited opportunity for such license, the Commission will give preferred consideration in the following order: First, to applications submitted by public or cooperative bodies for facilities to be located in high cost power areas in the United States; second, to applications submitted by others for facilities to be located in such areas; third, to applications submitted by public or cooperative bodies for facilities to be located in other than high cost power areas; and, fourth, to all other applicants. [21 FR 355]

(c) The licensee who transmits electric energy in interstate commerce, or sells it at wholesale in interstate commerce, shall be subject to the regulatory provisions of the Federal Power Act.

(d) Nothing herein shall preclude any government agency, now or hereafter authorized by law to engage in the production, marketing, or distribution of electric energy, if otherwise qualified, from obtaining a license for the construction and operation of a utilization facility for the primary purpose of producing electric energy for disposition for ultimate public consumption.

[1] As permitted by subsection 105c(8) of the Act, with respect to proceedings in which an application for a construction permit was filed prior to Dec. 19, 1970, and proceedings in which a written request for antitrust review of an application for an operating license to be issued under section 104b has been made by a person who intervened or sought by timely written notice to the Atomic Energy Commission to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination within 25 days after the date of publication in the FEDERAL REGISTER of notice of filing of the application for an operating license or Dec. 19, 1970, whichever is later, the Commission may issue a construction permit or operating license in advance of consideration of, and findings with respect to the antitrust aspects of the application, provided that the permit or license so issued contains the condition specified in § 50.55b.

§ 50.44 Standards for combustible gas control system in light water cooled power reactors.

(a) Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, shall, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by (1) metal-water reaction involving the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the reactor coolant, and (3) corrosion of metals.

(b) Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding shall be provided with the capability for (1) measuring the hydrogen concentration in the containment, (2) insuring a mixed atmosphere in the containment, and (3) controlling combustible gas concentrations in the containment following a postulated LOCA.

43 FR 50162

(c)(1) For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (i) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (ii) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function.

(2) If the conditions set out in paragraph (c)(1) of this section cannot be shown, the containment shall be provided with an inerted or an oxygen deficient atmosphere in order to provide protection against hydrogen burning and explosions during the time period specified in paragraph (c)(1) of this section.

(3) Notwithstanding paragraphs (c)(1) and (c)(2) of this section:

(i) Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment; and

46 FR 58484

(ii) By the end of the first scheduled outage beginning after July 5, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor that relies upon a purge/repressurization system as the primary means for controlling combustible gases following a LOCA shall be provided with either an internal recombiner or the capability to install an external recombiner following the start of an accident. The internal or external recombiners must meet the combustible gas control requirements in paragraph (d) of this section. The containment penetrations used for external recombiners must either be:

(A) dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures for containment isolation purposes, and be sized to satisfy the flow requirements of the external recombiners, or

(B) of a combined design for use by either external recombiners or purge/repressurization systems and other systems, conform to the requirements of criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/repressurization systems, and be sized to satisfy the flow requirements of the external recombiners or purge repressurization systems.

46 FR 58484

(iii) To provide improved operational capability to maintain adequate core cooling following an accident, by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. (High point vents are not required, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.

50 FR 3498

(iv) (A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

50 FR 3498

(B) Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on. Another method could include a showing that the following specific criteria of the ASME Boiler and Pressure Vessel Code are met:

(*1*) That steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in paragraph 50.55a(b)(1) of this part), specifically in Section III, Division 1, Subsubarticle NE–3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required); and

(*2*) That concrete containments meet the requirements of the ASME Boiler and Pressure vessel Code, Section III, Division 2, Subsubarticle CC–3720, Factored Load Category, considering pressure and dead load alone.

53 FR 43419

(C) Subsubarticle NE–3220, Division 1, and Subsubarticle CC–3720, Division 2, of Section III of the ASME Boiler and Pressure Vessel Code, referenced in paragraphs (c)(5)(iv)(B)(*1*) and (c)(3)(iv)(B)(*2*) of this section, have been approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L Street N.W., Washington, D.C.

50 FR 3498

(D) If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure:

(*1*) During the accident, where it is acceptable to show that it does not exceed Service Level C Limits or the Factored Load Category (as described in paragraph (c)(3)(iv)(B) of this section; and

(*2*) Following inadvertent full inerting during normal plant operations, where it

is acceptable to show that it does not exceed either the Service Level A Limits of Subsubarticle NE-3220 (for a steel containment) or the Service Load Category of Subsubarticle CC-3720 (for a concrete containment.)

(3) Modest deviations from the criteria in paragraph (c)(3)(iv)(D) of this section will be considered by the Commission if good cause is shown.

(E) If the hydrogen control system relies on post-accident inerting, the systems and components required to establish and maintain safe shutdown and containment integrity must be designed and qualified for the environment caused by such inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant.

(v) (A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall provide its nuclear power reactor with systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity. These systems and components must be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such determinations can be shown unlikely to occur.

(B) The amount of hydrogen to be considered is equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(vi)(A) Each applicant for or holder of an operating license for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall submit an analysis to the Commission as specified in § 50.4.

(B) The analysis required by paragraph (c)(3)(vi)(A) of this section must:

(1) Provide an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75% of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate;

(2) Include the period of recovery from the degraded condition;

(3) Use accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core.

(4) Support the design of the hydrogen control system selected under paragraph (c)(3)(iv) of this section; and,

(5) Show that for those reactors described in paragraph (c)(3)(iv) of this section that do not rely upon an inerted atmosphere to control hydrogen inside the containment:

(i) The containment structural integrity as described in paragraph (c)(3)(iv) of this section will be maintained; and

(ii) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such determinations can be shown unlikely to occur.

(vii)(A) By June 25, 1985, each applicant for or holder of an operating license subject to the requirements of paragraphs (c)(3)(iv), (v) and (vi) of this section shall develop and submit to the Commission a proposed schedule for meeting these requirements. The schedule may be developed using integrated scheduling systems previously approved for the facility by the NRC.

(B) For each applicant for an operating license as of February 25, 1985 the schedule shall provide for compliance with the requirements of paragraph (c)(3)(iv)(A) of this section prior to operation of the reactor in excess of 5 percent power. Completed final analyses are not necessary for a staff determination that a plant is safe to operate at full power provided that prior to such operation an applicant has provided a preliminary analysis which the staff has determined provides a satisfactory basis for a decision to support interim operation at full power until the final analysis has been completed. However, the record in this rulemaking shows that such preliminary analyses are not necessary for a staff determination that a plant is safe to operate at full power if the staff has determined for similar plants, referenced in this notice of rulemaking, that similar systems provide a satisfactory basis for a decision to support operation at full power until the preliminary analyses have been completed.

(C) For those holders of operating licenses containing license conditions on Hydrogen Control Measures covered by this section, the schedule shall be consistent with those license conditions, or approved amendments thereto.

(D) For those facilities not having an NRC approved integrated scheduling system, a final schedule for meeting the requirements of paragraphs (c)(3) (iv), (v) and (vi) of this section shall be established by the NRC staff within 90 days of receipt of a proposed schedule from the licensee or applicant, taking into account the current status of efforts at the facility to comply with the requirements; analyses that may be provided by applicants or licensees regarding the impacts of these requirements on other scheduled plant modifications, including any NRC-mandated safety modifications, and their relative importance to safety; and the Commission's objective that these requirements be complied with without undue delay.

(d)(1) For facilities that are in compliance with § 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with § 50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch (0.0058 mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs.

(2) For facilities as to which no evaluation of compliance in accordance with § 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reaction of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

(e) For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for controlled purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas con-

trol system, such as recombiners, that does not result in a singificant release from containment.

(f) For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970, if the incremental radiation dose from purging (and repressurization if a repressurization system is provided) occurring at all points beyond the exclusion area boundary after a postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 2.5 rem to the whole body and less than 30 rem to the thyroid, and if the combined radiation dose at the low population zone outer boundary from purging and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(g) For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise, the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(h) As used in this section:

(1) Degradation, but not total failure, of emergency core cooling functioning means that the performance of the emergency core cooling system is postulated, for purposes of design of the combustible gas control system, not to meet the acceptance criteria in § 50.46 and that there could be localized clad melting and metal-water reaction to the extent postulated in paragraph (d) of this section. The degree of performance degradation is not postulated to be sufficient to cause core meltdown.

(2) A combustible gas control system is a system that operates after a LOCA to maintain the concentrations of combustible gases within the containment, such as hydrogen, below flammability limits. Combustible gas control systems are of two types: (i) Systems that allow controlled release from containment, through filters if necessary, such as purging systems and repressurization systems, and (ii) systems that do not result in a significant release from containment such as recombiners.

(3) A purging system is a system for the controlled release of the containment atmosphere to the environment through filters if needed.

(4) A repressurization system is a system used to dilute the concentration of combustible gas within containment by adding inert gas or air to the containment. Dilution of the combustible gas results in a delay in time until a flammable concentration is reached and permits fission product decay. Operation is limited to a containment repressurization to 50 percent of the containment design pressure. A purging system is normally part of the repressurization system.

[43 FR 55162]

[26 FR 355]

§ 50.45 Standards for construction permits.

An applicant for a license or an amendment of a license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit, if the application is in conformity with and acceptable under the criteria of §§ 50.31 through 50.38 and the standards of §§ 50.40 through 50.43.

[53 FR 35996]

§ 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(a)(1)(i) Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding must be provided with an emergency core cooling system (ECCS) that must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, Required Documentation, sets forth the documentation requirements for each evaluation model.

(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K, ECCS Evaluation Models.

(2) The Director of Nuclear Reactor Regulations may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1)(i) and (ii) of this section.

[53 FR 35996]

(3)(i) Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

(ii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4. If the change or error is signficant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to

show compliance with § 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§ 50.55(e), 50.72 and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46 requirements.

(b)(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

(2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry.* Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(c) As used in this section:

(1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(2) An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A.

§ 50.47 Emergency plans.

(a)(1) Except as provided in paragraph (d) of this section, no operating license for a nuclear power reactor will be issued unless a finding is made by NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

(2) The NRC will base its finding on a review of the Federal Emergency Management Agency [FEMA] findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented. A FEMA finding will primarily be based on a review of the plans. Any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that the plans can be implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on questions of adequacy and implementation capability.

(b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

(3) Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

(5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

(6) Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

(7) Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

(9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

(10) A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

(11) Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

(12) Arrangements are made for medical services for contaminated injured individuals.

(13) General plans for recovery and reentry are developed.

(14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

(15) Radiological emergency response training is provided to those who may be called on to assist in an emergency.

(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

(c)(1) Failure to meet the applicable standards set forth in paragraph (b) of this section may result in the Commission declining to issue an operating license; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operations. Where an applicant for an operating license asserts that its inability to demonstrate compliance with the requirements of paragraph (b) of this section results wholly or substantially from the decision of state and/or local governments not to participate further in emergency planning, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction that:

(i) The applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of the non-participation of state and/or local governments.

(ii) The applicant has made a sustained, good faith effort to secure and retain the participation of the pertinent state and/or local governmental authorities, including the furnishing of copies of its emergency plan.

(iii) The applicant's emergency plan provides reasonable assurance that public health and safety is not endangered by operation of the facility concerned. To make that finding, the applicant must demonstrate that, as outlined below, adequate protective measures can and will be taken in the event of an emergency. A utility plan will be evaluated against the same planning standards applicable to a state or local plan, as listed in paragraph (b) of this section, with due allowance made both for—

(A) Those elements for which state and/or local non-participation makes compliance infeasible and

(B) The utility's measures designed to compensate for any deficiencies resulting from state and/or local non-participation.

In making its determination on the adequacy of a utility plan, the NRC will recognize the reality that in an actual emergency, state and local government officials will exercise their best efforts to protect the health and safety of the public. The NRC will determine the adequacy of that expected response, in combination with the utility's compensating measures, on a case-by-case basis, subject to the following guidance. In addressing the circumstance where applicant's inability to comply with the requirements of paragraph (b) of this section is wholly or substantially the result of non-participation of state and/or local governments, it may be presumed that in the event of an actual radiological emergency state and local officials would generally follow the utility plan. However, this presumption may be rebutted by, for example, a good faith and timely proffer of an adequate and feasible state and/or local radiological emergency plan that would in fact be relied upon in a radiological emergency.

(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

(d) Notwithstanding the requirements of paragraphs (a) and (b) of this section, and except as specified by this paragraph, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and capability to implement State and local or utility offsite emergency plans are required prior to issuance of an operating license authorizing only fuel loading or low power testing and training (up to 5 percent of the rated power). Insofar as emergency planning and preparedness requirements are concerned, a license authorizing fuel loading and/or low power testing and training may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The NRC will base this finding on its assessment of the applicant's onsite emergency plans against the pertinent standards in paragraph (b) of this section and Appendix E. Review of applicant's emergency plans will include the following standards with offsite aspects:

(1) Arrangements for requesting and effectively using offsite assistance on site have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned onsite response have been identified.

(2) Procedures have been established for licensee communications with State and local response organizations, including initial notification of the declaration of emergency and periodic provision of plant and response status reports.

(3) Provisions exist for prompt communications among principal response organizations to offsite emergency personnel who would be responding onsite.

(4) Adequate emergency facilities and equipment to support the emergency response onsite are provided and maintained.

(5) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use onsite.

(6) Arrangements are made for medical services for contaminated and injured onsite individuals.

(7) Radiological emergency response training has been made available to those offsite who may be called to assist in an emergency onsite.

53 FR 30865

PART 50 ● DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

§ 50.48 Fire protection.

(a) Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of Appendix A of this part. This fire protection plan must describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan must also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.[3] The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the facility license and shall retain each superseded revision of the procedures for three years from the date it was superseded.

(b) Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate prior to January 1, 1979. Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1[4] reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position BTP APCSB 9.5-1 was published in August 1976. With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate prior to January 1, 1979 shall satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.

(c) All fire protection modifications required to satisfy the provisions of Appendix R to this part or directly affected by such requirements shall be completed on the following schedule:

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training, shall be implemented within 30 days after the effective date of this section and Appendix R to this part.

(2) Those fire protection features that involve installation of modifications that do not require prior NRC approval or plant shutdown shall be implemented within 9 months after the effective date of this section and Appendix R to this part.

(3) Those fire protection features, except for those requiring prior NRC approval by paragraph (c)(5) of this section, that involve installation of modifications that do require plant shutdown, the need for which is justified in the plans and schedules required by the provisions of paragraph (c)(5) of this section, shall be implemented before startup after the earliest of the following events commencing 180 days or more after the effective date of this section and Appendix R to this part:

(i) the first refueling outage;

(ii) another planned outage that lasts for at least 60 days; or

(iii) an unplanned outage that lasts for at least 120 days.

(4) Those fire protection features that require prior NRC approval by paragraph (c)(5) of this section, shall be implemented within the following schedule: Dedicated shutdown systems—30 months after NRC approval; modifications requiring plant shutdown—before startup after the earliest of the events given in paragraph (c)(3) commencing 180 days after NRC approval; modifications not requiring plant shutdown—6 months after NRC approval.

(5) Licensees shall make any modifications necessary to comply with these requirements in accordance with the above schedule without prior review and approval by NRC except for modifications required by Section III.G.3 of Appendix R to this part. Licensees shall submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and Appendix R to this part. Licensees shall submit design descriptions of modifications needed to satisfy Section III.G.3 of Appendix R to this part within 30 days after the the effective date of this section and Appendix R to this part.

(6) In the event that a request for exemption from a requirement to comply with one or more of the provisions of Appendix R filed within 30 days of the effective date of this rule is based on an assertion by the licensee that such required modifications would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety, the schedule requirements of paragraph (c) shall be tolled until final Commission action on the exemption request upon a determination by the Director of Nuclear Reactor Regulation that the licensee has provided a sound technical basis for such assertion that warrants further staff review of the request.

(d) Fire protection features accepted by the NRC staff in Fire Protection Safety Evaluation Reports referred to in paragraph (b) of this section and supplements to such reports, other than features covered by paragraph (c), shall be completed as soon as practicable but no later than the completion date currently specified in license conditions or technical specifications for such facility, or the date determined by paragraphs (d)(1) through (d)(4) of this section, whichever is sooner, unless the Director of Nuclear Reactor Regulation determines, upon a showing by the licensee, that there is good cause for extending such date and that the public health and safety is not adversely affected by such extension. Extensions of such date shall not exceed the dates determined by paragraphs (c)(1) through (c)(4) of this section.

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training shall be implemented within 4 months after the date of the NRC staff Fire Protection Evaluation Report accepting or requiring such features.

(2) Those fire protection features involving installation of modifications not requiring prior approval or plant shutdown shall be implemented within 12 months after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features.

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[3] Basic fire protection guidance for nuclear power plants is contained in two NRC documents:

• Branch Technical Position Auxiliary Power Conversion System Branch BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," for new plants docketed after July 1, 1976, dated May 1976.

• Appendix A to BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976," for plants that were operating or under various stages of design or construction before July 1, 1976, dated August 23, 1976. Also see Note 4.

[4] Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5-1 has been provided in four other NRC documents:

• "Supplementary Guidance on Information Needed for Fire Protection Evaluation," dated October 21, 1976.

• "Sample Technical Specification," dated May 12, 1977.

• "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance," dated June 14, 1977.

• "Manpower Requirements for Operating Reactors," dated May 11, 1978.

A Fire Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of Appendix R to this part.

(3) Those fire protection features, including alternative shutdown capability, involving installation of modifications requiring plant shutdown shall be implemented before the startup after the earliest of the following events commencing 9 months or more after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days; or

(iii) An unplanned outage that lasts for at least 120 days.

(4) Those fire protection features involving dedicated shutdown capability requiring new buildings and systems shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.

(e) Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of Appendix A to this part in accordance with the provisions of their licenses.

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment:[4] This equipment is that relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment.[5]

(c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safety meets the requirements of paragraph (j) of this section.

(1) The performance specifications under conditions existing during and following design basis accidents.

(2) The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

(e) The electric equipment qualification program must include and be based on the following:

(1) *Temperature and Pressure.* The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

(2) *Humidity.* Humidity during design basis accidents must be considered.

(3) *Chemical Effects.* The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) *Radiation.* The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) *Aging.* Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.

(6) *Submergence* (if subject to being submerged).

(7) *Synergistic Effects.* Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

(8) *Margins.* Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

(f) Each item of electric equipment important to safety must be qualified by one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment

[4] Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE Std 308-1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, NY 10017.

[5] Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Copies of the Regulatory Guide may be purchased through the U.S. Government Printing Office by calling 202-275-2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082.

reference temperature as adjusted for the effects of neutron radiation for the period of service in question.

(5) "RT_{PTS}" means the reference temperature calculated by the method given in paragraph (b)(2) of this section for use as a screening criterion.

(b) *Requirements.*

(1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT_{PTS} (at the inner vessel surface) of reactor vessel beltline materials by giving values from the time of submittal to the expiration date of the operating license. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. This assessment must be submitted by January 23, 1986, and must be updated when changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values.

(2) The pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, or 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be calculated as follows. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline. For each material, RT_{PTS} is the lower of the results given by Equations 1 and 2.

Equation 1: $RT_{PTS} = I + M + [-10 + 470\ Cu + 350\ Cu\ Ni]\ f^{0.28}$

Equation 2: $RT_{PTS} = I + M + 283\ f^{0.28}$

(i) "I" means the initial reference temperature of the unirradiated material measured as defined in the ASME Code, Paragraph NB–2331. If a measured value is not available, the following generic mean values must be used: 0 °F for welds made with Linde 80 flux, and −56 °F for welds made with Linde 0091, 1092 and 124 and ARCOS B–5 weld fluxes.

(ii) "M" means the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence and the calculational procedures. In Equation 1, M = 48 °F if a measured value of I was used, and M = 59 °F if the generic mean value of I was used. In Equation 2, M = 0 °F if a measured value of I was used, and M = 34 °F if the generic mean value of I was used.

(iii) "Cu" and "Ni" mean the best estimate weight percent of copper and nickel in the material. The source of these values must be included in the assessment. The relationship of the material, on which any measurements were made, to the actual material in the pressure vessel must be described.

(iv) "f" means the best estimate neutron fluence, in unit of 10^{19} n/cm^2 (E greater than or equal to 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question.

(3) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, the licensee shall submit by April 23, 1986, an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2). The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated Commission approval of detailed plant-specific analyses, submitted to demonstrate acceptable risk at values of RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(4) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expiration date of the operating license, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine reactor vessel materials properties based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least 3 years before the value of RT_{PTS} as defined in paragraph (b)(2) of this section is projected to exceed the PTS screening criterion or by one year after issuance of the Commission Guidance and Acceptance Criteria for these analyses, whichever is later.

(5) After consideration of the licensee's analyses (including effects of proposed corrective actions, if any) submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Commission may, on a case-by-case basis, approve operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion. The Commission will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(6) If the Commission concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility at value of RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee must request and receive Commission approval prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

[50 FR 29937]

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants.

(b) *Definition.* For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be

[50 FR 29937]

[49 FR 26036]

program description included or referenced in the Safety Analysis Report, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC at least annually in accordance with the requirements of § 50.71. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval prior to implementation, as follows: 51 FR 43709

(i) Changes made to the Safety Analysis Report must be submitted, as specified in § 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted, as specified in § 50.4.

(ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of Appendix B of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items). 48 FR 1076

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(iv) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

(b) No right to the special nuclear material shall be conferred by the license except as may be defined by the license. 21 FR 355

(c) Neither the license, nor any right thereunder, nor any right to utilize or produce special nuclear material shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accordance with the provisions of the act and give its consent in writing. 21 FR 355

(d) The license shall be subject to suspension and to the rights of recapture of the material or control of the facility reserved to the Commission under section 108 of the act in a state of war or national emergency declared by Congress.

(e) The license shall be subject to revocation, suspension, modification, or amendment for cause as provided in the act and regulations, in accordance with the procedures provided by the act and regulations.

(f) The licensee shall at any time before expiration of the license, upon request of the Commission, submit, as specified in § 50.4, written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked.

Except for information sought to verify licensee compliance with the current licensing basis for that facility, the NRC must prepare the reason or reasons for each information request prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each such justification provided for an evaluation performed by the NRC staff must be approved by the Executive Director for Operations or his or her designee prior to issuance of the request. 52 FR 49363

(g) The issuance or existence of the license shall not be deemed to waive, or relieve the licensee from compliance with, the antitrust laws, as specified in subsection 105a of the Act. In the event that the licensee should be found by a court of competent jurisdiction to have violated any provision of such antitrust laws in the conduct of the licensed activity, the Commission may suspend or revoke the license or take such other action with respect to it as shall be deemed necessary. 21 FR 355

(h) The license shall be subject to the provisions of the Act now or hereafter in effect and to all rules, regulations, and orders of the Commission. The terms and conditions of the license shall be subject to amendment, revision, or modification, by reason of amendments of the Act or by reason of rules, regulations, and orders issued in accordance with the terms of the act.

(i) Except as provided in § 55.13 of this chapter, the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator as provided in Part 55 of this chapter. 52 FR 9453

(i-1) Within three months after issuance of an operating license, the licensee shall have in effect an operator requalification program which must as a minimum, meet the requirements of § 55.59(c) of this chapter. Notwithstanding the provisions of § 50.59, the licensee may not, except as specifically authorized by the Commission decrease the scope of an approved operator requalification program.

(j) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to Part 55 of this chapter present at the controls. 28 FR 3156

(k) An operator or senior operator licensed pursuant to Part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

(l) The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. These individuals shall be licensed as senior operators pursuant to Part 55 of this chapter.

(m)(1)A senior operator licensed pursuant to Part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license. 48 FR 31611

(2) Notwithstanding any other provisions of this section, by January 1, 1984, licensees of nuclear power units shall meet the following requirements:

(i) Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

(g) Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification.

The schedule in this paragraph supersedes the June 30, 1982, deadline, or any other previously imposed date, for environmental qualification of electric equipment contained in certain nuclear power operating licenses.

(h) Each license shall notify the Commission as specified in § 50.4 of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.

(i) Applicants for operating licenses granted after February 22, 1983, but prior to November 30, 1988, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted, as specified in § 50.4, for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

(1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.

(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power unit or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section—

(1) Is qualified for its application; and

(2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines), or NUREG–0588 (For Comment version), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

(l) Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.

Issuance, Limitations, and Conditions of Licenses and Construction Permits

§ 50.50 Issuance of licenses and construction permits.

Upon determination that an application for a license meets the standards and requirements of the act and regulations, and that notifications, if any, to other agencies or bodies have been duly made, the Commission will issue a license, or if appropriate a construction permit, in such form and containing such conditions and limitations including technical specifications, as it deems appropriate and necessary.

§ 50.51 Duration of license; renewal.

Each license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from the date of issuance. Where the operation of a facility is involved the Commission will issue the license for the term requested by the applicant or for the estimated useful life of the facility if the Commission determines that the estimated useful life is less than the term requested. Where construction of a facility is involved, the Commission may specify in the construction permit the period for which the license will be issued if approved pursuant to § 50.56. Licenses may be renewed by the Commission upon the expiration of the period. Application for termination of license is to be made pursuant to § 50.82.

§ 50.52 Combining licenses.

The Commission may combine in a single license the activities of an applicant which would otherwise be licensed severally.

§ 50.53 Jurisdictional limitations.

No license under this part shall be deemed to have been issued for activities which are not under or within the jurisdiction of the United States.

§ 50.54 Conditions of licenses.

Whether stated therein or not, the following shall be deemed conditions in every license issued:

(a)(1) Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in Appendix B of this part shall implement, pursuant to § 50.34(b)(6)(ii) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report.

(2) Each licensee described in paragraph (a)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a licensee need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.)

(3) After March 11, 1983, each licensee described in paragraph (a)(1) of this section may make a change to a previously accepted quality assurance

designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.

(d) *Implementation.* By 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988*, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

(c) *Implementation.*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant licensed to operate on or before *July 21, 1988*, the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by *April 17, 1989*. For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(2) *Alternate ac source.* The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. Alternate ac source(s) serving a multiple unit site where onsite emergency ac sources are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites where onsite emergency ac sources are shared between units, the alternate ac source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in § 50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required.

(3) *Regulatory Assessment:* After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section.

(4) *Implementation Schedule:* For each light-water-cooled nuclear power plant licensed to operate on or before June 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with paragraph (c)(3) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section will be established by the NRC staff in consultation and coordination with the affected licensee.

§ 50.64 Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.

(a) *Applicability.* The requirements of this section apply to all non-power reactors.

(b) *Requirements.* (1) The Commission will not issue a construction permit after March 27, 1986 for a non-power reactor where the applicant proposes to use highly enriched uranium (HEU) fuel, unless the applicant demonstrates that the proposed reactor will have a unique purpose as defined in § 50.2.

MINIMUM REQUIREMENTS [1] PER SHIFT FOR ON-SITE STAFFING OF NUCLEAR POWER UNITS BY OPERATORS AND SENIOR OPERATORS LICENSED UNDER 10 CFR PART 55

Number of nuclear power units operating [2]	Position	One unit	Two units		Three units	
		One control room	One control room	Two control rooms	Two control rooms	Three control rooms
None	Senior Operator	1	1	1	1	1
	Operator	1	2	2	3	3
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
Two	Senior Operator		2	3	[3] 3	3
	Operator		3	4	[3] 5	5
Three	Senior Operator				3	4
	Operator				5	6

Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.

[1] For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.

[2] The number of required licensed personnel when the operating nuclear power units are controlled from a common control room are two senior operators and four operators.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

(3) Licensees who cannot meet the January 1, 1984 deadline must submit by October 1, 1983 a request for an extension to the Director of the Office of Nuclear Regulation and demonstrate good cause for the request.

46 FR 31611

(n) The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or construction permit pursuant to § 50.36 of this part.

31 FR 3668

(o) Primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to this part.

36 FR 4385

(p)(1) The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with Appendix C of Part 73 of this chapter for effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may make no change which would decrease the effectiveness of a security plan, or guard training and qualification plan, prepared pursuant to § 50.34(c) or Part 73 of this chapter, or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, Responsibility Matrix) contained in a licensee safeguards contingency plan prepared pursuant to § 50.34(d) or Part 73 of this chapter, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to the licensee's license pursuant to § 50.90.

52 FR 1415

(2) The licensee may make changes to the plans referenced in paragraph (p)(1) without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of three years from the date of the change, and shall submit, as specified in § 50.4, a report containing a description of each change within two months after the change is made. Prior to the safeguards contingency plan being put into effect, the licensee shall have:

53 FR 19240

(i) All safeguards capabilities specified in the safeguards contingency plan available and functional,

(ii) Detailed procedures developed according to Appendix C to Part 73 available at the licensee's site, and

(iii) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed Procedures.

51 FR 47206

(3) The licensee shall provide for the development, revision, implementation, and maintanence of its safeguards contingency plan. To this end, the licensee shall provide for a review at least every 12 months of the safeguards contingency plan by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program. The review must include a review and audit of safeguards contingency procedures and practices, an audit of the security system testing and maintenance program, and a test of the safeguards systems along with commitments established for response by local law enforcement authorities. The results of

the review and audit, along with recommendations for improvements, must be documented, reported to the licensee's corporate and plant management, and kept available at the plant for inspection for a period of two years. [51 FR 47206]

(q) A licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in § 50.47(b) and the requirements in Appendix E of this part. A licensee authorized to possess and/or operate a research reactor or a fuel facility shall follow and maintain in effect emergency plans which meet the requirements in Appendix E to this part. The licensee shall retain the emergency plan and each change that decreases the effectiveness of the plan as a record until the Commission terminates the license for the nuclear power reactor. The nuclear power reactor licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of § 50.47(b) and the requirements of Appendix E to this part. The research reactor and/or the fuel facility licensee may make changes to these plans without Commission approval only if these changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the requirements of Appendix E to this part. This nuclear power reactor, research reactor, or fuel facility licensee shall retain a record of each change to the emergency plan made without prior Commission approval for a period of three years from the date of the change. Proposed changes that decrease the effectiveness of the approved emergency plans may not be implemented without application to and approval by the Commission. The licensee shall submit, as specified in § 50.4, a report of each proposed change for approval. If a change is made without approval, the licensee shall submit, as specified in § 50.4, a report of each change within 30 days after the change is made. [53 FR 19240]

(r) Each licensee who is authorized to possess and/or operate a research or test reactor facility with an authorized power level greater than or equal to 2 MW thermal, under a licensee of the type specified in § 50.21(c), shall submit emergency plans complying with 10 CFR Part 50, Appendix E, to the Director of the Office of Nuclear Reactor Regulation for approval by September 7, 1982. Each licensee who is authorized to possess and/or operate a research or test reactor facility with an authorized power level less than 2 MW thermal, under a license of the type specified in § 50.21(c), shall submit emergency plans complying with 10 CFR Part 50, Appendix E, to the Director of the Office of Nuclear Reactor Regulation for approval by November 8, 1982. [47 FR 19512]

(s)(1) Each licensee who is authorized to possess and/or operate a nuclear power reactor shall submit to NRC within 60 days of the effective date of this amendment the radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within a plume exposure pathway EPZ, as well as the plans of State governments wholly or partially within an ingestion pathway EPZ.[1] [2] [47 FR 41336]

Ten (10) copies of the above plans shall be forwarded to the Director of Nuclear Reactor Regulation with 3 copies to the Administrator of the appropriate NRC regional office. Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs for a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway EPZ shall focus on such actions as are appropriate to protect the food ingestion pathway. [47 FR 41336]

(2)(i) For operating power reactors, the licensee, State, and local emergency response plans shall be implemented by April 1, 1981, except as provided in Section IV.D.3 of Appendix E to this part.

(ii) If after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency (*including findings based on requirements of Appendix E, Section IV.D.3*) and if the deficiencies (*including deficiencies based on requirements of Appendix E, Section IV.D.3*) are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown or other enforcement action is appropriate, the Commission shall take into account, among other factors, whether the licensee can demonstrate to the Commission's satisfaction that the deficiencies in the plan are not significant for the plant in question, or that adequate interim compensating actions have been or will be taken promptly, or that that there are other compelling reasons for continued operation. [46 FR 63031]

[1] Emergency Planning Zones (EPZs) are discussed in NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978.

[2] If the State and local emergency response plans have been previously provided to the NRC for inclusion in the facility docket, the applicant need only provide the appropriate reference to meet this requirement.

(3) The NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the licensee's emergency plans are adequate and capable of being implemented. Nothing in this paragraph shall be construed as limiting the authority of the Commission to take action under any other regulation or authority of the Commission or at any time other than that specified in this paragraph.

(t) A nuclear power reactor licensee shall provide for the development, revision, implementation, and maintenance of its emergency preparedness program. To this end, the licensee shall provide for a review of its emergency preparedness program at least every 12 months by persons who have no direct responsibility for implementation of the emergency preparedness program. The review shall include an evaluation for adequacy of interfaces with State and local governments and of licensee drills, exercises, capabilities, and procedures. The results of the review, along with recommendations for improvements, shall be documented, reported to the licensee's corporate and plant management, and retained for a period of five years. The part of the review involving the evaluation for adequacy of interface with State and local governments shall be available to the appropriate State and local governments.

(u) Within 60 days after the effective date of this amendment, each nuclear power reactor licensee shall submit to the NRC plans for coping with emergencies that meet standards in § 50.47(b) and the requirements of Appendix E to this part.

45 FR 55402

(v) Each licensee subject to the requirements of Part 73 of this chapter shall ensure that physical security, safeguards contingency and guard qualification and training plans and other related Safeguards Information are protected against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate.

46 FR 51718

(w) Each electric utility licensee under this part for a production or utilization facility of the type described in § 50.21(b) or § 50.22 shall take reasonable steps to obtain insurance available at reasonable costs and on reasonable terms from private sources or to demonstrate to the satisfaction of the Commission that it possesses an equivalent amount of protection covering the licensee's obligation, in the event of an accident at the licensee's reactor, to stabilize and decontaminate the reactor and the reactor station site at which the reactor experiencing the accident is located, provided that:

55 FR 12163

(1) The insurance required by paragraph (w) of this section must have a minimum coverage limit for each reactor station site of either $1.06 billion or whatever amount of insurance is generally available from private sources, whichever is less. The required insurance must clearly state that, as and to the extent provided in paragraph (w)(4) of this section, any proceeds must be payable first for stabilization of the reactor and next for decontamination of the reactor and the reactor station site. If a licensee's coverage falls below the required minimum, the licensee shall within 60 days take all reasonable steps to restore its coverage to the required minimum. The required insurance may, at the option of the licensee, be included within policies that also provide coverage for other risks, including, but not limited to, the risk of direct physical damage.

(2)(i) With respect to policies issued or annually renewed on or after April 2, 1991, the proceeds of such required insurance must be dedicated, as and to the extent provided in this paragraph, to reimbursement or payment on behalf of the insured of reasonable expenses incurred or estimated to be incurred by the licensee in taking action to fulfill the licensee's obligation, in the event of an accident at the licensee's reactor, to ensure that the reactor is in, or is returned to, and maintained in, a safe and stable condition and that radioactive contamination is removed or controlled such that personnel exposures are consistent with the occupational exposure limits in 10 CFR part 20. These actions must be consistent with any other obligation the licensee may have under this chapter and must be subject to paragraph (w)(4) of this section. As used in this section, an "accident" means an event that involves the release of radioactive material from its intended place of confinement within the reactor or on the reactor station site such that there is a present danger of release off site in amounts that would pose a threat to the public health and safety.

55 FR 12163

(ii) The stabilization and decontamination requirements set forth in paragraph (w)(4) of this section must apply uniformly to all insurance policies required under paragraph (w) of this section.

(3) The licensee shall report to the NRC on April 1 of each year the current levels of this insurance or financial security it maintains and the sources of this insurance or financial security.

(4)(i) In the event of an accident at the licensee's reactor, whenever the estimated costs of stabilizing the licensed reactor and of decontaminating the reactor and the reactor station site exceed $100 million, the proceeds of the insurance required by paragraph (w) of this section must be dedicated to and used, first, to ensure that the licensed reactor is in, or is returned to, and can be maintained in, a safe and stable condition so as to prevent any significant risk to the public health and safety and, second, to decontaminate the reactor and the reactor station site in accordance with the licensee's cleanup plan as approved by order of the Director of the Office of Nuclear Reactor Regulation. This priority on insurance proceeds must remain in effect for 60 days or, upon order of the Director, for such longer periods, in increments not to exceed 60 days except as provided for activities under the cleanup plan required in paragraphs (w)(4)(iii) and (w)(4)(iv) of this section, as the Director may find necessary to protect the public health and safety. Actions needed to bring the reactor to and maintain the reactor in a safe and stable condition may include one or more of the following, as appropriate: (A) Shutdown of the reactor; (B) Establishment and maintenance of long-term cooling with stable decay heat removal; (C) Maintenance of subcriticality; (D) Control of radioactive releases; and (E) Securing of structures, systems, or components to minimize radiation exposure to onsite personnel or to the offsite public or to facilitate later decontamination or both.

55 FR 12163

(ii) The licensee shall inform the Director of the Office of Nuclear Reactor Regulation in writing when the reactor is and can be maintained in a safe and stable condition so as to prevent any significant risk to the public health and safety. Within 30 days after the licensee informs the Director that the reactor is in this condition, or at such earlier time as the licensee may elect or the Director may for good cause direct, the licensee shall prepare and submit a cleanup plan for the Director's approval. The cleanup plan must identify and contain an estimate of the cost of each cleanup operation that will be required to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under § 50.82 for authority to decommission the reactor and to surrender the license voluntarily. Cleanup operations may include one or more of the following, as appropriate: (A) Processing any contaminated water generated by the accident and by decontamination operations to remove radioactive materials; (B) Decontamination of surfaces inside the auxiliary and fuel-handling buildings and the reactor building to levels consistent with the Commission's occupational exposure limits in 10 CFR part 20, and decontamination or disposal of equipment; (C) Decontamination or removal and disposal of internal parts and damaged fuel from the reactor vessel; and (D) Cleanup of the reactor coolant system.

(iii) Following review of the licensee's cleanup plan, the Director will order the

licensee to complete all operations that the Director finds are necessary to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under § 50.82 for authority to decommission the reactor and to surrender the license voluntarily. The Director shall approve or disapprove, in whole or in part for stated reasons, the licensee's estimate of cleanup costs for such operations. Such order may not be effective for more than 1 year, at which time it may be renewed. Each subsequent renewal order, if imposed, may be effective for not more than 6 months.

(iv) Of the balance of the proceeds of the required insurance not already expended to place the reactor in a safe and stable condition pursuant to paragraph (w)(2)(i) of this section, an amount sufficient to cover the expenses of completion of those decontamination operations that are the subject of the Director's order shall be dedicated to such use, provided that, upon certification to the Director of the amounts expended previously and from time to time for stabilization and decontamination and upon further certification to the Director as to the sufficiency of the dedicated amount remaining, policies of insurance may provide for payment to the licensee or other loss payees of amounts not so dedicated, and the licensee may proceed to use in parallel (and not in preference thereto) any insurance proceeds not so dedicated for other purposes.

[55 FR 1216]

(x) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.

(y) Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator prior to taking the action.

[48 FR 13946]

(z) Each licensee with a utilization facility licensed pursuant to sections 103 or 104b. of the Act shall immediately notify the NRC Operations Center of the occurrence of any event specified in § 50.72 of this part.

[48 FR 39039]

(aa) The license shall be subject to all conditions deemed imposed as a matter of law by sections 401(a)(2) and 401(d) of the Federal Water Pollution Control Act, as amended (33 U.S.C.A. 1341(a)(2) and (d).)

[49 FR 9352]

(bb) For operating nuclear power reactors, the licensee shall, no later than 5 years before expiration of the reactor operating license, submit written notification to the Commission for its review and preliminary approval of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor upon expiration of the reactor operating license until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository. Final Commission review will be undertaken as part of any proceeding for continued licensing under Part 50 or Part 72. The licensee must demonstrate to NRC that the elected actions will be consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented on a timely basis. Where implementation of such actions require NRC authorizations, the licensee shall verify in the notification that submittals for such actions have been or will be made to NRC and shall identify them. A copy of the notification shall be retained by the licensee as a record until expiration of the reactor operating license. The licensee shall notify the NRC of any significant changes in the proposed waste management program as described in the initial notification.

[49 FR 34688]

(cc)(1) Each licensee shall notify the appropriate NRC Regional Administrator, in writing, immediately following the filing of a voluntary or involuntary petition for bankruptcy under any Chapter of Title 11 (Bankruptcy) of the United States Code by or against:

(i) The licensee;

(ii) An entity (as that term is defined in 11 U.S.C. 101(14)) controlling the licensee or listing the license or licensee as property of the estate; or

(iii) An affiliate (as that term is defined in 11 U.S.C. 101(2)) of the licensee.

(2) This notification must indicate:

(i) The bankruptcy court in which the petition for bankruptcy was filed; and

(ii) The date of the filing of the petition.

[52 FR 1292]

(dd) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in a national security emergency:

(1) When this action is immediately needed to implement national security objectives as designated by the national command authority through the Commission, and

(2) No action consistent with license conditions and technical specifications that can meet national security objectives is immediately apparent.

[54 FR 7178]

A national security emergency is established by a law enacted by the Congress or by an order or directive issued by the President pursuant to statutes or the Constitution of the United States. The authority under this paragraph must be exercised in accordance with law, including section 57e of the Act, and is in addition to the authority granted under paragraph (x) of this section, which remains in effect unless otherwise directed by the Commission during a national security emergency.

[54 FR 7178]

§ 50.55 Conditions of construction permits.

Each construction permit shall be subject to the following terms and conditions:

(a) The permit shall state the earliest and latest dates for completion of the construction or modification.

(b) If the proposed construction or modification of the facility is not completed by the latest completion date, the permit shall expire and all rights thereunder shall be forfeited: *Provided, however,* That upon good cause shown the Commission will extend the completion date for a reasonable period of time. The Commission will recognize, among other things, developmental problems attributable to the experimental nature of the facility or fire, flood, explosion, strike, sabotage, domestic violence, enemy action, an act of the elements, and other acts beyond the control of the permit holder, as a basis for extending the completion date.

[42 FR 43385]

(c) Except as modified by this section and § 50.55a, the construction permit shall be subject to the same conditions to which a license is subject.

[36 FR 11423]

(d) At or about the time of completion of the construction or modification of the facility, the applicant will file any additional information needed to bring the original application for license up to date, and will file an application for an operating license or an amendment to an application for a license to construct and operate the facility for the issuance of an operating license, as appropriate, as specified in § 50.36(d) of this part.

[35 FR 19555]

(e) (1) If the permit is for construction of a nuclear power plant, the holder of the permit shall notify the Commission of each deficiency found in design and construction, which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant, and which represents:

(i) A significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B to this part; or

[37 FR 6459]

(ii) A significant deficiency in final design as approved and released for construction such that the design does not conform to the criteria and bases stated in the safety analysis report or construction permit; or

(iii) A significant deficiency in construction of or significant damage to a structure, system, or component which will require extensive evaluation, extensive redesign, or extensive repair to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function; or

(iv) A significant deviation from performance specifications which will require extensive evaluation, extensive redesign, or extensive repair to establish the adequacy of a structure, system, or component to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function.

(2) The holder of a construction permit shall within 24 hours notify the appropriate Nuclear Regulatory Commission Regional Office of each reportable deficiency.

(3)(i) The holder of a construction permit shall also submit, as specified in § 50.4, a written report on a reportable deficiency within 30 days.

(ii) The report must include a description of the deficiency, an analysis of the safety implications and the corrective action taken, and sufficient information to permit analysis and evaluation of the deficiency and of the corrective action. If sufficient information is not available for a definitive report to be submitted within 30 days, an interim report containing all available information shall be filed, as specified in § 50.4, together with a statement that indicates when a complete report will be filed.

(4) Remedial action may be taken both prior to and after notification of the Division of Inspection and Enforcement subject to the risk of subsequent disapproval of such action by the Commission.

(f)(1) Each nuclear power plant or fuel reprocessing plant construction permit holder subject to the quality assurance criteria in Appendix B of this part shall implement, pursuant to § 50.34(a)(7) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report.

(2) Each construction permit holder described in paragraph (f)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a permit holder need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.)

(3) After March 11, 1983, each construction permit holder described in paragraph (f)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to NRC within 90 days. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval before implementation, as follows:

(i) Changes to the Safety Analysis Report must be submitted for review as specified in § 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in § 50.4.

(ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of Appendix B of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items).

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(iv) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

§ 50.55a Codes and standards.

Each operating license for a boiling or pressurized water-cooled nuclear power facility shall be subject to the conditions in paragraph (g) of this section and each construction permit for a utilization facility shall be subject to the following conditions in addition to those specified in § 50.55.

(a)(1) Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(2) Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.

(3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant must demonstrate that (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

(b) The ASME Boiler and Pressure Vessel Code, which is referenced in the following paragraphs, was approved for incorporation by reference by the Director of the Federal Register

A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L St. NW., Washington, D.C.

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1983 Edition and Addenda through the Winter 1985 Addenda, and the 1986 Edition.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1, and include editions through the 1983 Edition and Addenda through the Winter 1985 Addenda, and the 1986 Edition, subject to the following limitations and modifications:

(i) *"Limitations on specific editions and addenda."* When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations. [46 FR 63288]

(ii) *Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB-2500 and IWB-2500-1 and Category B-J).* If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB-2500 and Table IWB-2600 Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt. [44 FR 57913]

(iii) *Steam generator tubing (modifies Article IWB-2000).* If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications.

(iv) Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC-2520 or IWC-2520-1, Category C-F). (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems, shall be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of Section XI of the ASME Code, the extent of examination for these systems shall be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda. [53 FR 16051]

(B) For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of Section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

(v) Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB-3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB-3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB-3640, as modified by the Winter 1985 Addenda. [53 FR 16051]

(c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III [1], [2] of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section. [49 FR 9711]

(2) Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, Provided: [50 FR 34085]

(i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or

(ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) The Code Edition, Addenda, and optional Code Cases [3] to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition, (iii) the ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition, and (iv) ASME Code Cases [4] must have been determined suitable for use by the NRC. [49 FR 9711]

(4) For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit. [49 FR 19857]

(d) *Quality Group B components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B [5] must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code Edition, Addenda, and optional Code Cases [4] to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler Vessel and Pressure Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases [4] must have been determined suitable for use by the NRC.

(e) *Quality Group C components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group C [5] must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code. [49 FR 9711]

(2) The Code Edition, Addenda, and optional Code Cases [4] to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA-1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases [4] must have been determined suitable for use by the NRC.

(f) [reserved] 49 FR 9711

See footnotes at end of section.

49 FR 9711

(g) Inservice inspection requirements:

50 FR 38970

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves shall meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda * in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

41 FR 22931

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

50 FR 38970

(i) Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda* applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda * applied to the construction of the particular component.

(iii) Pumps and valves which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda* applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the Boiler and Pressure Vessel Code and Addenda * applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

44 FR 57912

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

41 FR 6256

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

44 FR 57912

(i) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during the initial 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety, and system pressure tests, conducted during successive 120-month inspection intervals shall comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inpsection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved] 50 FR 38970

See footnotes at end of section.

44 FR 57932

(iv) Inservice examinations of components, tests of pumps and valves, and system pressure tests, may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

41 FR 21931

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility shall be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

51 FR 40303

(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall submit this application, as specified in § 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by paragraph (g)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.

[Reserved]

(footnotes 2 and 3) [reserved]

[1] USAS and ASME Code addenda issued prior to the Winter 1977 Addenda are considered to be "in effect" or "effective" 6 months after their date of issuance and after they are incorporated by reference in paragraph (b) of this section. Addenda to the ASME Code issued after the Summer 1977 Addenda are considered to be "in effect" or "effective" after the date of publication of the addenda and after they are incorporated by reference in paragraph (b) of this section.

[1] 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.

[1] ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability—ASME Section III Division 1," NRC Regulatory Guide 1.85, "Materials Code Case Acceptability—ASME Section III Division 1," and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI Division 1." The use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3). 53 FR 16051

[1] 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 Commission's Technical Library, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland.

[1] Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a-1) and Subpart F of Part 2 of this chapter, "the formal docket date of the application for a construction permit" for purposes of this section shall be the date of docketing of the information required by § 2.101(a-1)(2) or (3), whichever is later.

[1] Guidance for quality group classifications of components which are to be included in the safety analysis reports pursuant to § 50.34(a) and 50.34(b) may be found in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants," and in Section 3.2.2 of NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants."

41 FR 6256

(iv) Where an examination or test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination or test is determined to be impractical.

43 FR 56035

(6)(i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

41 FR 6256

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

42 FR 22783

(h) Protection systems. For construction permits issued after January 1, 1971, protection systems shall meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE-279) in effect[1] on the formal docket date[1] of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.

50 FR 38809

(i) [removed] 50 FR 38950

49 FR 9711

(j) [reserved] 49 FR 9711

[1] Amended 48 FR 5532

[50.55b Revoked 43 FR 49775

§ 50.56 Conversion of construction permit to license; or amendment of license.

Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be. [35 FR 11461]

§ 50.57 Issuance of operating license.[1] [36 FR 5317]

(a) Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and

(4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter. However, no finding of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in § 50.21(b) or § 50.22. [49 FR 35747]

(5) The applicable provisions of Part 140 of this chapter have been satisfied; and

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public. [36 FR 5317]

(b) Each operating license will include appropriate provisions with respect to any uncompleted items of construction and such limitations or conditions as are required to assure that operation during the period of the completion of such items will not endanger public health and safety.

(c) An applicant may, in a case where a hearing is held in connection with a pending proceeding under this section make a motion in writing, pursuant to this paragraph (c), for an operating license authorizing low-power testing (operation at not more than 1 percent of full power for the purpose of testing the facility), and further operations short of full power operation. Action on such a motion by the presiding officer shall be taken with due regard to the rights of the parties to the proceeding, including the right of any party to be heard to the extent that his contentions are relevant to the activity to be authorized. Prior to taking any action on such a motion which any party opposes, the presiding officer shall make findings on the matters specified in paragraph (a) of this section as to which there is a controversy, in the form of an initial decision with respect to the contested activity sought to be authorized. The Director of Nuclear Reactor Regulation will make findings on all other matters specified in paragraph (a) of this section. If no party opposes the motion, the presiding officer will issue an order pursuant to § 2.730(e) of this chapter, authorizing the Director of Nuclear Reactor Regulation to make appropriate findings on the matters specified in paragraph (a) of this section and to issue a license for the requested operation. [37 FR 15127]

(d) [Removed, 51 FR 7744]

[1] The Commission may issue a provisional operating license pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional operating license or a notice of proposed issuance of a provisional operating license has been published on or before that date.

§ 50.58 Hearings and report of the Advisory Committee on Reactor Safeguards.

(a) Each application for a construction permit or an operating license for a facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility, shall be referred to the Advisory Committee on Reactor Safeguards for a review and report. An application for an amendment to such a construction permit or operating license may be referred to the Advisory Committee on Reactor Safeguards for review and report. Any report shall be made part of the record of the application and available to the public, except to the extent that security classification prevents disclosure.

(b)(1) The Commission will hold a hearing after at least 30-days' notice and publication once in the Federal Register on each application for a construction permit for a production or utilization facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility.

(2) When a construction permit has been issued for such a facility following the holding of a public hearing, and an application is made for an operating license or for an amendment to a construction permit or operating license, the Commission may hold a hearing after at least 30-days' notice and publication once in the Federal Register, or, in the absence of a request therefor by any person whose interest may be affected, may issue an operating license or an amendment to a construction permit or operating license without a hearing, upon 30-days' notice and publication once in the Federal Register of its intent to do so.

(3) If the Commission finds, in an emergency situation, as defined in § 50.91, that no significant hazards consideration is presented by an application for an amendment to an operating license, it may dispense with public notice and comment and may issue the amendment. If the Commission finds that exigent circumstances exist, as described in § 50.91, it may reduce the period provided for public notice and comment.

(4) Both in an emergency situation and in the case of exigent circumstances, the Commission will provide 30 days notice of opportunity for a hearing, though this notice may be published after issuance of the amendment if the Commission determines that no significant hazards consideration is involved.

(5) The Commission will use the standards in § 50.92 to determine whether a significant hazards consideration is presented by an amendment to an operating license for a facility of the type described in § 50.21(b) or § 50.22, or which is a testing facility, and may make the amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

(6) No petition or other request for review of or hearing on the staff's significant hazards consideration determination will be entertained by the Commission. The staff's determination is final, subject only to the Commission's discretion, on its own initiative, to review the determination.

§ 50.59 Changes, tests and experiments.

(a)(1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b)(1) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the safety analysis report or to the extent that they constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records must include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

(2) The licensee shall submit, as specified in § 50.4, a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. The report must be submitted annually or at such shorter intervals as may be specified in the license.

(3) The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to § 50.90.

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

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

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.

§ 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

(a) *Definitions.* For the purposes of this section:

(1) "ASME Code" means the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda as specified by § 50.55a, Codes and Standards.

(2) "Pressurized Thermal Shock Event" means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

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

(4) "Initial RT_{NDT}" means the reference temperature for a reactor vessel material as defined in the ASME Code, Paragraph NB-2331. RT_{NDT} means the

reference temperature as adjusted for the effects of neutron radiation for the period of service in question.

(5) "RT_{PTS}" means the reference temperature calculated by the method given in paragraph (b)(2) of this section for use as a screening criterion.

(b) *Requirements.*

(1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT_{PTS} (at the inner vessel surface) of reactor vessel beltline materials by giving values from the time of submittal to the expiration date of the operating license. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. This assessment must be submitted by January 23, 1986, and must be updated whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values.

(2) The pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, or 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be calculated as follows. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline. For each material, RT_{PTS} is the lower of the results given by Equations 1 and 2.

Equation 1: $RT_{PTS} = I + M + [-10 + 470\ Cu + 350\ Cu\ Ni]\ f^{(0.28 - 0.10 \log f)}$

Equation 2: $RT_{PTS} = I + M + 283\ f^{(0.28 - 0.10 \log f)}$

(i) "I" means the initial reference temperature of the unirradiated material measured as defined in the ASME Code, Paragraph NB–2331. If a measured value is not available, the following generic mean values must be used: 0 °F for welds made with Linde 80 flux, and −56 °F for welds made with Linde 0091, 1092 and 124 and ARCOS B–5 weld fluxes.

(ii) "M" means the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence and the calculational procedures. In Equation 1, M = 48 °F if a measured value of I was used, and M = 59 °F if the generic mean value of I was used. In Equation 2, M = 0 °F if a measured value of I was used, and M = 34 °F if the generic mean value of I was used.

(iii) "Cu" and "Ni" mean the best estimate weight percent of copper and nickel in the material. The source of these values must be included in the assessment. The relationship of the material, on which any measurements were made, to the actual material in the pressure vessel must be described.

(iv) "f" means the best estimate neutron fluence, in unit of 10^{19} n/cm^2 (E greater than or equal to 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question.

(3) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, the licensee shall submit by April 23, 1986, an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2). The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated Commission approval of detailed plant-specific analyses, submitted to demonstrate acceptable risk at values of RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(4) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expiration date of the operating license, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine reactor vessel materials properties based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least 3 years before the value of RT_{PTS} as defined in paragraph (b)(2) of this section is projected to exceed the PTS screening criterion or by one year after issuance of the Commission Guidance and Acceptance Criteria for these analyses, whichever is later.

(5) After consideration of the licensee's analyses (including effects of proposed corrective actions, if any) submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Commission may, on a case-by-case basis, approve operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion. The Commission will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(6) If the Commission concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility at value of RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee must request and receive Commission approval prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

[50 FR 29937]

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants.

(b) *Definition.* For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be

[49 FR 26036]

designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device. [49 FR 26036]

[54 FR 13361] (4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

[49 FR 26036] (5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.

[51 FR 40303] (d) *Implementation.* By 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

[53 FR 23203]

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988,* if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

[53 FR 23203] (c) *Implementation.*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant licensed to operate on or before *July 21, 1988,* the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by *April 17, 1989.* For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(2) *Alternate ac source:* The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test. Alternate ac source(s) serving a multiple unit site where onsite emergency ac sources are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites where onsite emergency ac sources are shared between units, the alternate ac source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in § 50.2. If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required.

(3) *Regulatory Assessment:* After consideration of the information submitted in accordance with paragraph (c)(1) of this section, the Director, Office of Nuclear Reactor Regulation, will notify the licensee of the Director's conclusions regarding the adequacy of the proposed specified station blackout duration, the proposed equipment modifications and procedures, and the proposed schedule for implementing the procedures and modifications for compliance with paragraph (a) this section.

(4) *Implementation Schedule:* For each light-water-cooled nuclear power plant licensed to operate on or before June 21, 1988, the licensee shall, within 30 days of the notification provided in accordance with paragraph (c)(3) of this section, submit to the Director of the Office of Nuclear Reactor Regulation a schedule commitment for implementing any equipment and associated procedure modifications necessary to meet the requirements of paragraph (a) of this section. This submittal must include an explanation of the schedule and a justification if the schedule does not provide for completion of the modifications within two years of the notification provided in accordance with paragraph (c)(3) of this section. A final schedule for implementing modifications necessary to comply with the requirements of paragraph (a) of this section will be established by the NRC staff in consultation and coordination with the affected licensee.

[51 FR 6514]

§ 50.64 Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.

(a) *Applicability.* The requirements of this section apply to all non-power reactors.

(b) *Requirements.* (1) The Commission will not issue a construction permit after March 27, 1986 for a non-power reactor where the applicant proposes to use highly enriched uranium (HEU) fuel, unless the applicant demonstrates that the proposed reactor will have a unique purpose as defined in § 50.2.

(2) Unless the Commission has determined, based on a request submitted in accordance with paragraph (c)(1) of this section, that the non-power reactor has a unique purpose, each licensee authorized to possess and use HEU fuel in connection with the reactor's operation shall:

(i) Not initiate acquisition of additional HEU fuel, if low enriched uranium (LEU) fuel acceptable to the Commission for that reactor is available when it proposes that acquisition; and

(ii) Replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor, in accordance with a schedule determined pursuant to paragraph (c)(2) of this section.

(3) If not required by paragraphs (b)(1) and (2) of this section to use LEU fuel, the applicant or licensee must use HEU fuel of enrichment as close to 20% as is available and acceptable to the Commisson.

(c) *Implementation.* (1) Any request by a licensee for a determination that a non-power reactor has a unique purpose as defined in § 50.2 should be submitted with supporting documentation to the Director of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, by September 29, 1986.

(2) (i) By March 27, 1987 and at 12-month intervals thereafter, each licensee of a non-power reactor authorized to possess and use HEU fuel shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a written proposal for meeting the requirements of paragraph (b)(2) or (3) of this section. The licensee shall include in the proposal a certification that Federal Government funding for conversion is available through the Department of Energy (DOE) or other appropriate Federal Agency. The licensee shall also include in the proposal a schedule for conversion, based upon availability of replacement fuel acceptable to the Commisson for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for the available financial support, and reactor usage.

(ii) If Federal Government funding for conversion cannot be certified, the proposal's contents may be limited to a statement of this fact. If a statement of non-availability of Federal Government funding for conversion is submitted by a licensee, then it shall be required to resubmit a proposal for meeting the requirements of paragraph (b)(2) or (3) of this section at 12-month intervals.

(iii) The proposal shall include, to the extent required to effect the conversion, all necessary changes in the license, facility, or procedures. Supporting safety analyses should be provided so as to meet the schedule established for conversion. As long as Federal Government funding for conversion is not available, the resubmittal may be a reiteration of the original proposal. The Director of the Office of Nuclear Reactor Regulation shall review the proposal and confirm the status of Federal Government funding for conversion and, if a schedule for conversion has been submitted by the licensee, will then determine a final schedule.

(3) After review of the safety analysis required by paragraph (c)(2), the Director of the Office of Nuclear Reactor Regulation will issue an appropriate enforcement order directing both the conversion and, to the extent consistent with protection of the public health and safety, any necessary changes to the license, facility, or procedures.

§ 50.65 [Deleted 43 FR 6915.]

INSPECTIONS, RECORDS, REPORTS, NOTIFICATIONS

§ 50.70 Inspections.

(a) Each licensee and each holder of a construction permit shall permit inspection, by duly authorized representatives of the Commission, of his records, premises, activities, and of licensed materials in possession or use, related to the license or construction permit as may be necessary to effectuate the purposes of the Act, including section 105 of the Act.

(b)(1) Each licensee and each holder of a construction permit shall upon request by the Director, Office of Nuclear Reactor Regulation, provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee and each holder of a construction permit. The office shall be convenient to and have full access to the facility and shall provide the inspector both visual and acoustic privacy.

(2) For a site with a single power reactor or fuel facility licensed pursuant to Part 50, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 square feet either within the site's office complex or in an office trailer or other on site space is suggested as a guide. For sites containing multiple power reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspector(s). The office space that is provided shall be subject to the approval of the Director, Office of Nuclear Reactor Regulation. All furniture, supplies and communication equipment will be furnished by the Commission.

(3) The licensee or construction permit holder shall afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Administrator as likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection and personal safety.

(4) The licensee or construction permit holder (nuclear power reactor only) shall ensure that the arrival and presence of an NRC inspector, who has been properly authorized facility access as described in paragraph (b)(3) of this section, is not announced or otherwise communicated by its employees or contractors to other persons at the facility unless specifically requested by the NRC inspector.

§ 50.71 Maintenance of records, making of reports.

(a) Each licensee and each holder of a construction permit shall maintain all records and make all reports, in connection with the activity, as may be required by the conditions of the license or permit or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act, including section 105 of the Act. Reports must be submitted in accordance with § 50.4.

(b) With respect to any production or utilization facility of a type described in § 50.21(b) or 50.22, or a testing facility, each licensee and each holder of a construction permit shall submit its annual financial report, including the certified financial statements, to the Commission, as specified in § 50.4, upon issuance of the report.

(c) Records that are required by the regulations in this part, by license condition, or by technical specifications, must be retained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, these records must be retained until the Commission terminates the facility license.

(d)(1) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations. The record may also be stored in electronic media with the capability of producing legible, accurate, and complete records during the required retention period. Records such as letters, drawings, specifications, must include

[53 FR 19240] all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.

[41 FR 18300] (2) If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to § 50.12 of this part, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

[45 FR 30614] (e) Each person licensed to operate a nuclear power reactor pursuant to the provisions of § 50.21 or § 50.22 of this part shall update periodically, as provided in paragraphs (e)(3) and (4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.

[51 FR 40303] (1) The licensee shall submit revisions containing updated information to the Commission, as specified in § 50.4, on a replacement-page basis that is accompanied by a list which identifies the current pages of the FSAR following page replacement.

[45 FR 30614] (2) The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of § 50.59 but not previously submitted to the Commission.

[48 FR 40682] (3)(i) A revision of the original FSAR containing those original pages that are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

(ii) Not less than 15 days before § 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed. Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.

[47 FR 30614] (4) Subsequent revisions shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.

(5) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both).

[53 FR 19240] (6) The updated FSAR shall be retained by the licensee until the Commission terminates their license.

[48 FR 40682] [48 FR 39039]

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) *General Requirements.*[1] (1) Each nuclear power reactor licensee licensed under § 50.21(b) or § 50.22 of this part shall notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan;[2] or

(ii) Of those non-Emergency events specified in paragraph (b) of this section.

(2) If the Emergency Notification System is inoperative, the licensee shall make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Operations Center.[3]

[48 FR 39039] (3) The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.

(4) When making a report under paragraph (a)(3) of this section, the licensee shall identify:

(i) The Emergency Class declared; or

(ii) Either paragraph (b)(1), "One-Hour Report," or paragraph (b)(2), "Four-Hour Report," as the paragraph of this section requiring notification of the Non-Emergency Event.

(b) *Non-Emergency Events.* (1) *One-Hour Reports.* If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee shall notify the NRC as soon as practical and in all cases within one hour of the occurrence of any of the following:

(i)(A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.

[48 FR 40682] (ii) Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromises plant safety;

[48 FR 39039] (B) In a condition that is outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

[48 FR 40682] (iii) Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.

[48 FR 39039] (iv) Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal.

(v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

[1] Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular, § 20.205, § 20.403, § 50.36, and § 73.71.

[2] These Emergency Classes are addressed in Appendix E of this part.

[3] Commercial telephone number of the NRC Operations Center is (202) 951-0550.

(vi) Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases. [48 FR 40682]

(2) *Four-Hour Reports.* If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following: [48 FR 39039]

(i) Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety. [48 FR 40682]

(ii) Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported.

(iii) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: [48 FR 39039]

(A) Shut down the reactor and maintain it in a safe shutdown condition,

(B) Remove residual heat,

(C) Control the release of radioactive material, or

(D) Mitigate the consequences of an accident.

(iv)(A) Any airborne radioactive release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour.

(B) Any liquid effluent release that exceeds 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour. (Immediate notifications made under this paragraph also satisfy the requirements of paragraphs (a)(2) and (b)(2) of § 20.403 of Part 20 of this chapter.)

(v) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(vi) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials. [48 FR 39039]

(vii) Any instance of: [55 FR 29181]

(A) A defect in any spent fuel storage cask structure, system, or component which is important to safety; or

(B) A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask under a general license issued under § 72.210 of this chapter.

A followup written report is required by § 72.216(b) of this chapter including a description of the means employed to repair any defects or damage and prevent recurrence, using instructions in § 72.4, within 30 days of the report submitted in paragraph (a). A copy of the written report must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office shown in appendix D to part 20 of this chapter.

(c) *Followup Notification.* With respect to the telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, shall during the course of the event: [48 FR 39039]

(1) *Immediately report:* (i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.

(2) *Immediately report:* (i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

§ 50.73 Licensee event report system.

(a) *Reportable events.* (1) The holder of an operating license for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 30 days after the discovery of the event. Unless otherwise specified in this section, the licensee shall report an event regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

(2) The licensee shall report: [48 FR 33850]

(i)(A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications; or

(B) Any operation or condition prohibited by the plant's Technical Specifications; or

(C) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.

(ii) Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromised plant safety;

(B) In a condition that was outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

(iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

(iv) Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported.

(v) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design,

analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

(vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(viii)(A) Any airborne radioactivity release that exceeded 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour.

(B) Any liquid effluent release that exceeded 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.

(ix) Reports submitted to the Commission in accordance with paragraph (a)(2)(viii) of this section also meet the effluent release reporting requirements of paragraph 20.405(a)(5) of Part 20 of this chapter.

(x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(b) *Contents.* The Licensee Event Report shall contain:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2)(i) A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event.

(ii) The narrative description must include the following specific information as appropriate for the particular event:

(A) Plant operating conditions before the event.

(B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event.

(C) Dates and approximate times of occurrences.

(D) The cause of each component or system failure or personnel error, if known.

(E) The failure mode, mechanism, and effect of each failed component, if known.

(F) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER.

(1) The Energy Industry Identification System is defined in: IEEE Std 803–1983 (May 16, 1983) Recommended Practices for Unique Identification Plants and Related Facilities—Principles and Definitions.

(2) IEEE Std 803–1983 has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register**. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, NY 10017. IEEE Std 803–1983 is available for inspection at the NRC's Technical Library, which is located in the Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland; and at the Office of the Federal Register, 1100 L Street, NW, Washington, DC.

(G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected.

(H) For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.

(I) The method of discovery of each component or system failure or procedural error.

(J)*(1)* Operator actions that affected the course of the event, including operator errors, procedural deficiencies, or both, that contributed to the event.

(2) For each personnel error, the licensee shall discuss:

(i) Whether the error was a cognitive error (e.g., failure to recognize the actual plant condition, failure to realize which systems should be functioning, failure to recognize the true nature of the event) or a procedural error;

(ii) Whether the error was contrary to an approved procedure, was a direct result of an error in an approved procedure, or was associated with an activity or task that was not covered by an approved procedure;

(iii) Any unusual characteristics of the work location (e.g., heat, noise) that directly contributed to the error; and

(iv) The type of personnel involved (i.e., contractor personnel, utility-licensed operator, utility nonlicensed operator, other utility personnel).

(K) Automatically and manually initiated safety system responses.

(L) The manufacturer and model number (or other identification) of each component that failed during the event.

(3) An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.

(4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events at the same plant that are known to the licensee.

(6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.

(c) *Supplemental information.* The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee shall submit, as specified in § 50.4, the requested information as a supplement to the initial LER.

(d) *Submission of reports.* Licensee Event Reports must be prepared on Form NRC 366 and submitted within 30 days of discovery of a reportable event or situation to the U.S. Nuclear Regulatory Commission, as specified in § 50.4.

(e) *Report legibility.* The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.

(f) *Exemptions.* Upon written request from a licensee including adequate justification or at the initiation of the NRC staff, the NRC Executive Director for Operations may, by a letter to the licensee, grant exemptions to the reporting requirements under this section.

(g) *Reportable occurrences.* The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.

48 FR 33850
49 FR 47823
48 FR 33850
48 FR 33850
51 FR 40303
48 FR 33850

§ 50.74 Notification of change in operator or senior operator status.

Each licensee shall notify the Commission in accordance with § 50.4 within 30 days of the following in regard to a licensed operator or senior operator:

(a) Permanent reassignment from the position for which the licensee has certified the need for a licensed operator or senior operator under § 55.31(a)(3) of this chapter;

(b) Termination of any operator or senior operator;

(c) Disability or illness as described in § 55.25 of this chapter.

§ 50.75 Reporting and recordkeeping for decommissioning planning.

(a) This section establishes requirements for indicating to NRC how reasonable assurance will be provided that funds will be available for decommissioning. For electric utilities it consists of a step-wise procedure as provided in paragraphs (b), (c), (e), and (f) of this section. Funding for decommissioning of electric utilities is also subject to the regulation of agencies (e.g., Federal Energy Regulatory Commission (FERC) and State Public Utility Commissions) having jurisdiction over rate regulation. The requirements of this section, in particular paragraph (c), are in addition to, and not substitution for, other requirements, and are not intended to be used, by themselves, by other agencies to establish rates.

(b) Each electric utility applicant for or holder of an operating license for a production or utilization facility of the type and power level specified in paragraph (c) of this section shall submit a decommissioning report, as required by § 50.33(k) of this part containing a certification that financial assurance for decommissioning will be provided in an amount which may be more but not less than the amount stated in the table in paragraph (c)(1) of this section, adjusted annually using a rate at least equal to that stated in paragraph (c)(2) of this section, by one or more of the methods described in paragraph (e) of this section as acceptable to the Commission. The amount stated in the applicant's or licensee's certification may be based on a cost estimate for decommissioning the facility. As part of the certification, a copy of the financial instrument obtained to satisfy the requirements of paragraph (e) of this section is to be submitted to NRC.

(c) Table of minimum amounts (January 1986 dollars) required to demonstrate reasonable assurance of funds for decommissioning by reactor type and power level. P (in MWt); adjustment factor.[1]

	Millions
(1)(i) For a PWR:	
greater than or equal to 3400 MWt	$105
between 1200 MWt and 3400 MWt (For a PWR of less than 1200 MWt, use P = 1200 MWt)	$(75 + 0.0068P)
(ii) For a BWR:	
greater than or equal to 3400 MWt	$135
between 1200 MWt and 3400 MWt (For a BWR of less than 1200 MWt, use P = 1200 MWt)	$(104 + 0.009P)

(2) An adjustment factor at least equal to 0.65 L + 0.13 E + 0.22 B is to be used where L and E are escalation factors for labor and energy, respectively, and are to be taken from regional data of U.S. Department of Labor Bureau of Labor Statistics and B is an escalation factor for waste burial and is to be taken from NRC report NUREG–1307, "Report on Waste Burial Charges."

(d) Each non-electric utility applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by § 50.33(k) of this part containing a cost estimate for decommissioning the facility, an indication of which method or methods described in paragraph (e) of this section as acceptable to the Commission will be used to provide funds for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

(e)(1) As provided in paragraphs (e)(2) and (3) of this section, financial assurance is to be provided by the following methods:

(i) Prepayment. Prepayment is the deposit prior to the start of operation into an account segregated from licensee assets and outside the licensee's administrative control of cash or liquid assets such that the amount of funds would be sufficient to pay decommissioning costs. Prepayment may be in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities.

(ii) External sinking fund. An external sinking fund is a fund established and maintained by setting funds aside periodically in an account segregated from licensee assets and outside the licensee's administrative control in which the total amount of funds would be sufficient to pay decommissioning costs at the time termination of operation is expected. An external sinking fund may be in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities.

(iii) A surety method, insurance, or other guarantee method. These methods guarantee that decommissioning costs will be paid should the licensee default. A surety method may be in the form of a surety bond, letter of credit, or line of credit. Any surety method or insurance used to provide financial insurance for decommissioning must contain the following conditions:

(A) The surety method or insurance must be open-ended or, if written for a specified term, such as five years, must be renewed automatically unless 90 days or more prior to the renewal date, the issuer notifies the Commission, the beneficiary, and the licensee of its intention not to renew. The surety or insurance must also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if the licensee fails to provide a replacement acceptable to the Commission within 30 days after receipt of notification of cancellation.

(B) The surety or insurance must be payable to a trust established for decommissioning costs. The trustee and trust must be acceptable to the Commission. An acceptable trustee includes an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

(C) The surety method or insurance must remain in effect until the Commission has terminated the license.

(2) For a licensee other than an electric utility, acceptable methods of providing financial assurance for decommissioning are—

(i) Prepayment;

(ii) An external sinking fund, in which deposits are made at least annually, coupled with a surety method or insurance, the value of which may decrease by the amount being accumulated in the sinking fund.

(iii) A surety method, insurance, or other guarantee method. A parent company guarantee of funds for decommissioning costs based on a financial test may be used if the guarantee and test are as contained in Appendix A of 10 CFR Part 30. A parent company guarantee may not be used in combination with other financial methods to satisfy the requirements of this section.

(iv) In the case of Federal, State, or local government licensees, a statement of intent containing a cost estimate for decommissioning, and indicating that funds for decommissioning will be obtained when necessary.

[1] Amounts are based on activities related to the definition of "Decommission" in § 50.2 of this part and do not include the cost of removal and disposal of spent fuel or of nonradioactive structures and materials beyond that necessary to terminate the license.

53 FR 24018

53 FR 24018

53 FR 24018

(3) For an electric utility, acceptable methods of providing financial assurance for decommissioning are—

(i) Prepayment;

(ii) An external sinking fund in which deposits are made at least annually;

(iii) A surety method or insurance; and

(iv) In the case of Federal government licensees, a statement of intent containing a cost estimate for decommissioning or an amount based on paragraph (c) of this section, and indicating that funds for decommissioning will be obtained when necessary.

(f) Each licensee shall at or about 5 years prior to the projected end of operation submit a preliminary decommissioning plan containing a cost estimate for decommissioning and an up-to-date assessment of the major technical factors that could affect planning for decommissioning. Factors to be considered in submitting this information include—

(1) The decommissioning alternative anticipated to be used. The requirements of § 50.82(b)(1) must be considered at this time;

(2) Major technical actions necessary to carry out decommissioning safely;

(3) The current situation with regard to disposal of high-level and low-level radioactive waste;

(4) Residual radioactivity criteria;

(5) Other site specific factors which could affect decommissioning planning and cost.

If necessary, this submittal shall also include plans for adjusting levels of funds assured for decommissioning to demonstrate that a reasonable level of assurance will be provided that funds will be available when needed to cover the costs of decommissioning.

(g) Each licensee shall keep records of information important to the safe and effective decommissioning of the facility in an identified location until the license is terminated by the Commission. If records of relevant information are kept for other purposes, reference to these records and their locations may be used. Information the Commission considers important to decommissioning consists of—

(1) Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when significant contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations.

(2) As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. If required drawings are referenced, each relevant document need not be indexed individually. If drawings are not available, the licensee shall substitute appropriate records of available information concerning these areas and locations.

(3) Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

§ 50.78 Installation information and verification.

Each holder of a construction permit shall, if requested by the Commission, submit installation information on Form N-71, permit verification thereof by the International Atomic Energy Agency, and take such other action as may be necessary to implement the US/IAEA Safeguards Agreement, in the manner set forth in §§ 75.6 and 75.11 through 75.14 of this chapter.

TRANSFERS OF LICENSES—CREDITORS' RIGHTS—SURRENDER OF LICENSES

§ 50.80 Transfer of licenses.

(a) No license for a production or utilization facility, or any right thereunder, shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall give its consent in writing.

(b) An application for transfer of a license shall include as much of the information described in §§ 50.33 and 50.34 of this part with respect to the identity and technical and financial qualifications of the proposed transferee as would be required by those sections if the application were for an initial license, and, if the license to be issued is a class 103 license, the information required by § 50.33a. The Commission may require additional information such as data respecting proposed safeguards against hazards from radioactive materials and the applicant's qualifications to protect against such hazards. The application shall include also a statement of the purposes for which the transfer of the license is requested, the nature of the transaction necessitating or making desirable the transfer of the license, and an agreement to limit access to Restricted Data pursuant to § 50.37. The Commission may require any person who submits an application for license pursuant to the provisions of this section to file a written consent from the existing licensee or a certified copy of an order or judgment of a court of competent jurisdiction attesting to the person's right (subject to the licensing requirements of the Act and these regulations) to possession of the facility involved.

(c) After appropriate notice to interested persons, including the existing licensee, and observance of such procedures as may be required by the Act or regulations or orders of the Commission, the Commission will approve an application for the transfer of a license, if the Commission determines:

(1) That the proposed transferee is qualified to be the holder of the license; and

(2) That transfer of the license is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission pursuant thereto.

§ 50.81 Creditor regulations.

(a) Pursuant to section 184 of the Act, the Commission consents, without individual application, to the creation of any mortgage, pledge, or other lien upon any production or utilization facility not owned by the United States which is the subject of a license or upon any leasehold or other interest in such facility: *Provided*,

(1) That the rights of any creditor so secured may be exercised only in compliance with and subject to the same requirements and restrictions as would apply to the licensee pursuant to the provisions of the license, the Atomic Energy Act of 1954, as amended, and regulations issued by the Commission pursuant to said Act; and

(2) That no creditor so secured may take possession of the facility pursuant to the provisions of this section prior to either the issuance of a license from the Commission authorizing such possession or the transfer of the license.

(b) Any creditor so secured may apply for transfer of the license covering such facility by filing an application for transfer of the license pursuant to § 50.80(b). The Commission will act upon such application pursuant to § 50.80 (c).

(c) Nothing contained in this regulation shall be deemed to affect the means of acquiring, or the priority of, any tax lien or other lien provided by law.

(d) As used in this section:

(1) "License" includes any license or construction permit which may be issued by the Commission with regard to the facility;

(2) "Creditor" includes, without implied limitation, the trustee under any mortgage, pledge or lien on a facility made to secure any creditor, any trustee or receiver of the facility appointed by a court of competent jurisdiction in any action brought for the benefit of

any creditor secured by such mortgage, pledge or lien, any purchaser of such facility at the sale thereof upon foreclosure of such mortgage, pledge, or lien or upon exercise of any power of sale contained therein, or any assignee of any such purchaser.

53 FR 23356

§ 50.82 Application for termination of license.

(a) Any licensee may apply to the Commission for authority to surrender a license voluntarily and to decommission the facility. For a facility that permanently ceases operation after July 27, 1988, this application must be made within two years following permanent cessation of operations, and in no case later than one year prior to expiration of the operating license. Each application for termination of license must be accompanied, or preceded, by a proposed decommissioning plan. For a facility which has permanently ceased operation prior to July 27, 1988, requirements for contents of the decommissioning plan as specified in paragraphs (b) through (d) of this section may be modified with approval of the Commission to reflect the fact that the decommissioning process has been initiated previously.

(b) The proposed decommissioning plan must include—

(1) The choice of the alternative for decommissioning with a description of activities involved.

(i) For an electric utility licensee, an alternative is acceptable if it provides for completion of decommissioning within 60 years. Consideration will be given to an alternative which provides for completion of decommissioning beyond 60 years only when necessary to protect the public health and safety. Factors to be considered in evaluating an alternative which provides for completion of decommissioning beyond 60 years are set out in paragraph (b)(1)(iii) of this section.

(ii) For a licensee other than an electric utility, an alternative is acceptable if it provides for completion of decommissioning without significant delay. Consideration will be given to an alternative which provides for delayed completion of decommissioning only when necessary to protect the public health and safety. Factors to be considered in evaluating an alternative which provides for delayed completion of decommissioning are set out in paragraph (b)(1)(iii) of this section.

(iii) Factors to be considered in making the evaluations required by paragraphs (b)(1)(i) and (b)(1)(ii) of this section include unavailability of waste disposal capacity and other site specific factors affecting the licensee's capability to carry out decommissioning safely, including presence of other nuclear facilities at the site.

53 FR 24018

(2) A description of controls and limits on procedures and equipment to protect occupational and public health and safety.

(3) A description of the planned final radiation survey.

(4) An updated cost estimate for the chosen alternative for decommissioning, comparison of that estimate with present funds set aside for decommissioning, and plan for assuring the availability of adequate funds for completion of decommissioning.

(5) A description of technical specifications, quality assurance provisions and physical security plan provisions in place during decommissioning.

(c) Decommissioning plans which propose an alternative that delays completion of decommissioning by including a period of storage or long-term surveillance must provide that—

(1) Funds needed to complete decommissioning be placed into an account segregated from licensee assets and outside the licensee's administrative control during the storage or surveillance period, or a surety method or fund statement of intent be maintained in accordance with the criteria of § 50.75(e), and

(2) Means be included for adjusting cost estimates and associated funding levels over the storage or surveillance period.

(d) For decommissioning plans in which the major dismantlement activities are delayed by first placing the facility in storage, planning for those delayed activities may be less detailed. Updated detailed plans must be submitted and approved prior to the start of these activities.

(e) If the decommissioning plan demonstrates that the decommissioning will be performed in accordance with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public, and after notice to interested persons, the Commission will approve the plan subject to such conditions and limitations as it deems appropriate and necessary and issue an order authorizing the decommissioning.

(f) The Commission will terminate the license if it determines that—

(1) The decommissioning has been performed in accordance with the approved decommissioning plan and the order authorizing decommissioning, and

(2) The terminal radiation survey and associated documentation demonstrates that the facility and site are suitable for release for unrestricted use.

Amendment of License or Construction Permit at Request of Holder

§ 50.90 Application for amendment of license or construction permit.

Whenever a holder of a license or construction permit desires to amend the license or permit, application for an amendment must be filed with the Commission, as specified in § 50.4, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

51 FR 40303

§ 50.91 Notice for public comment; State consultation.

The Commission will use the following procedures on an application requesting an amendment to an operating licensee for a facility licensed under § 50.21(b) or § 50.22 or for a testing facility:

51 FR 40303

(a) *Notice for public comment.* (1) At the time a licensee requests an amendment, it must provide to the Commission, in accordance with the distribution requirements specified in § 50.4, its analysis about the issue of no significant hazards consideration using the standards in § 50.92.

51 FR 7744

(2)(i) The Commission may publish in the Federal Register under § 2.105 an individual notice of proposed action for an amendment for which it makes a proposed determination that no significant hazards consideration is involved, or, at least once every 30 days, publish a periodic Federal Register notice of proposed actions which identifies each amendment issued and each amendment proposed to be issued since the last such periodic notice, or it may publish both such notices.

(ii) For each amendment proposed to

be issued, the notice will (A) contain the staff's proposed determination, under the standards in § 50.92, (B) provide a brief description of the amendment and of the facility involved, (C) solicit public comments on the proposed determination, and (D) provide for a 30-day comment period.

(iii) The comment period will begin on the day after the date of the publication of the first notice, and, normally, the amendment will not be granted until after this comment period expires.

(3) The Commission may inform the public about the final disposition of an amendment request for which it has made a proposed determination of no significant hazards consideration either by issuing an individual notice of issuance under § 2.106 of this chapter or by publishing such a notice in its periodic system of **Federal Register** notices. In either event, it will not make and will not publish a final determination on no significant hazards consideration, unless it receives a request for a hearing on that amendment request.

(4) Where the Commission makes a final determination that no significant hazards consideration is involved and that the amendment should be issued, the amendment will be effective upon issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in § 2.714 of this chapter has filed a request for a hearing. The Commission need hold any required hearing only after it issues an amendment, unless it determines that a significant hazards consideration is involved in which case the Commission will provide an opportunity for a prior hearing.

(5) Where the Commission finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the Commission will not publish a notice of proposed determination on no significant hazards consideration, but will publish a notice of issuance under § 2.106 of this chapter, providing for opportunity for a hearing and for public comment after issuance. The Commission expects its licensees to apply for license amendments in timely fashion. It will decline to dispense with notice and comment on the determination of no significant hazards consideration if it determines that the licensee has abused the emergency provision by failing to make timely application for the amendment and thus itself creating the emergency. Whenever an emergency situation exists, a licensee requesting an amendment must explain why this emergency situation occurred and why it could not avoid this situation, and the Commission will assess the licensee's reasons for failing to file an application sufficiently in advance of that event.

(6) Where the Commission finds that exigent circumstances exist, in that a licensee and the Commission must act quickly and that time does not permit the Commission to publish a **Federal Register** notice allowing 30 days for prior public comment, and it also determines that the amendment involves no significant hazards considerations, it:

(i)(A) Will either issue a **Federal Register** notice providing notice of an opportunity for hearing and allowing at least two weeks from the date of the notice for prior public comment, or

(B) Will use local media to provide reasonable notice to the public in the area surrounding a licensee's facility of the licensee's amendment and of its proposed determination as described in paragraph (a)(2) of this section, consulting with the licensee on the proposed media release and on the geographical area of its coverage;

(ii) Will provide for a reasonable opportunity for the public to comment, using its best efforts to make available to the public whatever means of communication it can for the public to respond quickly, and, in the case of telephone comments, have these comments recorded or transcribed, as necessary and appropriate;

(iii) When it has issued a local media release, may inform the licensee of the public's comments, as necessary and appropriate;

(iv) Will publish a notice of issuance under § 2.106;

(v) Will provide a hearing after issuance, if one has been requested by a person who satisfies the provisions for intervention called for in § 2.714 of this chapter;

(vi) Will require the licensee to explain the exigency and why the licensee cannot avoid it, and use its normal public notice and comment procedures in paragraph (a)(2) of this section if it determines that the licensee has failed to use its best efforts to make a timely application for the amendment in order to create the exigency and to take advantage of this procedure.

(7) Where the Commission finds that significant hazards considerations are involved, it will issue a **Federal Register** notice providing an opportunity for a prior hearing even in an emergency situation, unless it finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR Part 2.

(b) *State consultation.*

(1) At the time a licensee requests an amendment, it must notify the State in which its facility is located of its request by providing that State with a copy of its application and its reasoned analysis about no significant hazards considerations and indicate on the application that it has done so. (The Commission will make available to the licensee the name of the appropriate State official designated to receive such amendments.)

(2) The Commission will advise the State of its proposed determination about no significant hazards consideration normally by sending it a copy of the **Federal Register** notice.

(3) The Commission will make available to the State official designated to consult with it about its proposed determination the names of the Project Manager or other NRC personnel it designated to consult with the State. The Commission will consider any comments of that State official. If it does not hear from the State in a timely manner, it will consider that the State has no interest in its determination; nonetheless, to ensure that the State is aware of the application, before it issues the amendment, it will make a good faith effort to telephone that official. (Inability to consult with a responsible State offical following good faith attempts will not prevent the Commission from making effective a license amendment involving no significant hazards consideration.)

(4) The Commission will make a good faith attempt to consult with the State before it issues a license amendment involving no significant hazards consideration. If, however, it does not have time to use its normal consultation procedures because of an emergency situation, it will attempt to telephone the appropriate State official. (Inability to consult with a responsible State official following good faith attempts will not prevent the Commission from making effective a license amendment involving no significant hazards consideration, if the Commission deems it necessary in an emergency situation.)

(5) After the Commission issues the requested amendment, it will send a copy of its determinaton to the State.

(c) *Caveats about State consultation.*

(1) The State consultation procedures in paragraph (b) of this section do not give the State a right:

(i) To veto the Commission's proposed or final determination;

(ii) To a hearing on the determination before the amendment becomes effective; or

(iii) To insist upon a postponement of the determination or upon issuance of the amendment.

(2) These procedures do not alter present provisions of law that reserve to the Commission exclusive responsibility for setting and enforcing radiological health and safety requirements for nuclear power plants.

§ 50.92 Issuance of amendment.

(a) In determining whether an amendment to a license or construction permit will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. If the application involves the material alteration of a licensed facility, a construction permit will be issued before the issuance of the amendment to the license. If the amendment involves a significant hazards consideration, the Commission will give notice of its proposed action (1) pursuant to § 2.105 of this chapter before acting thereon and (2) as soon as practicable after the application has been docketed.

(b) The Commission will be particularly sensitive to a license amendment request that involves irreversible consequences (such as one that permits a significant increase in the amount of effluents or radiation emitted by a nuclear power plant).

(c) The Commission may make a final determination, pursuant to the procedures in § 50.91, that a proposed amendment to an operating license for a facility licensed under § 50.21(b) or § 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

[51 FR 7744]

REVOCATION, SUSPENSION, MODIFICATION, AMENDMENT OF LICENSES AND CONSTRUCTION PERMITS, EMERGENCY OPERATIONS BY THE COMMISSION

§ 50.100 Revocation, suspension, modification of licenses and construction permits for cause.

A license or construction permit may be revoked, suspended, or modified, in whole or in part, for any material false statement in the application for license or in the supplemental or other statement of fact required of the applicant; or because of conditions revealed by the application for license or statement of fact or any report, record, inspection, or other means, which would warrant the Commission to refuse to grant a license on an original application (other than those relating to §§ 50.51, 50.42(a), and 50.43(b) of this part); or for failure to construct or operate a facility in accordance with the terms of the construction permit or license, provided that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit shall be governed by the provisions of § 50.55(b); or for violation of, or failure to observe, any of the terms and provisions of the act, regulations, license, permit, or order of the Commission.

[21 FR 355]

§ 50.101 Retaking possession of special nuclear material.

Upon revocation of a license, the Commission may immediately cause the retaking of possession of all special nuclear material held by the licensee.

§ 50.102 Commission order for operation after revocation.

Whenever the Commission finds that the public convenience and necessity, or the Department finds that the production program of the Department requires continued operation of a production or utilization facility, the license for which has been revoked, the Commission may, after consultation with the appropriate federal or state regulatory agency having jurisdiction, order that possession be taken of such facility and that it be operated for a period of time as, in the judgment of the Commission, the public convenience and necessity or the production program of the Department may require, or until a license for operation of the facility shall become effective. Just compensation shall be paid for the use of the facility.

[45 FR 14199]

§ 50.103 Suspension and operation in war or national emergency.

[21 FR 355]

(a) Whenever Congress declares that a state of war or national emergency exists, the Commission, if it finds it necessary to the common defense and security, may,

(1) Suspend any license it has issued.

[40 FR 8747]

(2) Cause the recapture of special nuclear material.

[21 FR 355]

(3) Order the operation of any licensed facility.

(4) Order entry into any plant or facility in order to recapture special nuclear material or to operate the facility.

(b) Just compensation shall be paid for any damages caused by recapture of special nuclear material or by operation of any facility, pursuant to this section.

[54 FR 20603]

§ 50.109 Backfitting.

(a)(1) Backfitting is defined as the 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 or 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 staff position after:

(i) The date of issuance of the construction permit for the facility for facilities having construction permits issued after October 21, 1985; or

(ii) Six months before the date of docketing of the operating license application for the facility for facilities having construction permits issued before October 21, 1985; or

[53 FR 20603]

(iii) The date of issuance of the operating license for the facility for facilities having operating licenses; or

[54FR15372]

(iv) The date of issuance of the design approval under Appendix M, N, or O of Part 52.

(2) Except as provided in paragraph (a)(4) of this section, the Commission shall require a systematic and documented analysis pursuant to paragraph (c) of this section for backfits which it seeks to impose.

(3) Except as provided in paragraph (a)(4) of this section, the Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

[53 FR 20603]

(4) The provisions of paragraphs (a)(2) and (a)(3) of this section are inapplicable and, therefore, backfit analysis is not required and the standards in paragraph (a)(3) of this section do not apply where the Commission or staff, as appropriate, finds and declares, with appropriated documented evaluation for its finding, either:

(i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or

(ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or

(iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate.

(5) The Commission shall always require the backfitting of a facility if it determines that such regulatory action is

adequate assurance that Nuclear Power Plant (NPP) structures, systems, and components will satisfactorily perform their safety functions. Appendix B of 10 CFR Part 50 establishes QA requirements for the design, construction, and operation of those structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Regulatory Guide 1.33 has been promulgated to describe acceptable methods for complying with the provisions of these appendices of 10 CFR Part 50 for the operational phase of NPPs.

These requirements are directed towards maintaining a quality assurance program and do not explicitly address maintenance as defined in the proposed rule. The proposed Maintenance Standard and Regulatory Guide will directly address the maintenance activities in the proposed rule.

7. The estimated resource burden on the NRC associated with the backfit and the availability of such resources.

The estimated total cost for NRC review of industry submittals required by the proposed maintenance rule is $400,000 based on the need for developing a Regulatory Guide to provide the basis for review of an industry-developed Maintenance Standard or to provide guidance on implementation of the rule if a Maintenance Standard is not developed. Individual plant maintenance programs will be subject to the ongoing inspection process but would not be required to be submitted to the NRC for review and approval.

8. The potential impact of difference in facility type design, or age on the relevancy and practicality of the backfit.

The proposed maintenance rule establishes generic requirements applicable to all types of facilities and designs, regardless of their age.

9. Whether the backfit is interim or final and, if interim, the justification for imposing the backfit on an interim basis.

The proposed maintenance rule would be a final requirement.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this proposed regulation will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed regulation affects licensees that own and operate nuclear utilization facilities licensed under sections 103 and 104 of the Atomic Energy Act of 1954, as amended. These licensees do not fall within the definition of small business set forth in section 3 of the Small Business Act, 15 U.S.C. 632, or within the Small Business Size Standard set forth in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Nuclear power plants and reactors.

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, Secs. 234, 83 Stat. 1244, 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). Sections 50.10 also issued under Secs. 101, 185, 68, Stat. 936, 955, as amended (42 U.S.C. 4332). 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). Sections 50.103 also issued under Sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2237).

For the purposes of Sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a) (b), and (c) 50.44, 50.46, 50.48 and 50.54 and 50.54(a) are issued under Sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c) and 50.54 are issued under Sec. 161i, 68, Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.9, 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73 and 50.78 are issued under Sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. A new § 50.65 is added to read as follows:

§ 50.65 Requirements to ensure the effectiveness of maintenance programs for nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all nuclear power reactors licensed under § 50.21 b or § 50.22.

(b) *Definitions.* For the purpose of this section, the Commission defines maintenance as the aggregate of those planned and systematic actions required to preserve and promptly restore operability, reliability, and availability of, or to prevent the failure of, plant structures, systems, and components. The Commission intends the scope of the rule to cover all systems, structures and components, including those in the Balance-of-Plant. Maintenance includes not only activities traditionally associated with identifying and correcting actual or potential degraded condition, i.e., repair, surveillance, diagnostic examinations, and preventive measures; but extends to all supporting functions for the conduct of these activities. Maintenance includes the following activities:

(1) Technology in the areas of:

(i) Corrective maintenance,

(ii) Preventive maintenance,

(iii) Predictive maintenance, and

(iv) Maintenance Surveillance;

(2) Engineering in support of maintenance;

(3) Quality assurance and quality control of maintenance activities;

(4) Incorporation of plant modification into the maintenance program;

(5) Equipment history and trending;

(6) Maintenance record keeping;

(7) Management of parts, tools, and facilities;

(8) Maintenance procedures;

(9) Post-Maintenance testing and return-to-service activities;

(10) Measures of overall maintenance program effectiveness;

(11) Maintenance management and organization in the areas of:

(i) Planning,

(ii) Scheduling,

(iii) Staffing,

(iv) Shift coverage, and

(v) Resource allocation;

(12) Control of contracted maintenance services;

(13) Radiological exposure control (including ALARA) during maintenance activities;

(14) Maintenance personnel qualification and training;

(15) Internal communications between the maintenance organization and plant operations and support groups;

(16) Communications between plant and corporate management and the maintenance organization; and

(17) Consideration of maintenance recommendations or requirements of individual vendors.

(c) *Requirements.* Each holder of an operating license subject to this section shall (1) establish, implement, and maintain an effective and documented maintenance program that addresses elements and activities in paragraph (b) of this section, and (2) regularly assess the effectiveness of this maintenance program and, based upon this assessment, make improvements, as appropriate.

(d) *Implementation.* By (insert a date 2 years after the effective date of the amendment) each licensee shall certify, by letter to the Director of the Office of

Nuclear Reactor Regulation, that a comprehensive documented maintenance program is being maintained and implemented, which addresses all elements and activities in paragraph (b) of this section including measures to monitor the effectiveness of the maintenance program and to improve the program where appropriate. In addition, each licensee shall develop (insert a date 3 months after the effective date of the amendment) a timely and expeditious plan and schedule (including Key Milestones) for meeting the requirements of this section.

Dated at Rockville, Maryland, this 21st day of November, 1988.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

53 FR 49[illegible]
Published 12/13/88

10 CFR Part 50

Public Meeting on Notice of Proposed Rulemaking on Maintenance of Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: On November 28, 1988 (53 FR 47822), the Commission published a notice of proposed rulemaking on maintenance of nuclear power plants. This meeting is scheduled as a result of a request from Nuclear Management and Resources Council (NUMARC). The agenda for the meeting will mainly include time for NRC staff responses to questions from NUMARC on the notice of proposed rulemaking including the Commission's solicitation of proposals for the development of a maintenance standard.

DATE: 1:00 p.m., December 14, 1988.

ADDRESS: Room 4B11, One White Flint North, 11155 Rockville Pike, Rockville, MD 20852.

FOR FURTHER INFORMATION CONTACT: Moni Dey, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 492-3730.

Dated in Rockville, Maryland this 6th day of December, 1988.

For the Nuclear Regulatory Commission.

Moni Dey,

Task Manager, Advanced Reactors & Generic Issues Branch, Division of Regulatory Applications, Office of Nuclear Regulatory Research.

53 FR 52716
Published 12/29/88
Comment period extended to 2/27/89.

10 CFR Part 50

Ensuring the Effectiveness of Maintenance Programs for Nuclear Power Plants; Extension of Comment Period

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule. Extension of comment period.

SUMMARY: On November 28, 1988 (53 FR 47822) the Commission published for public comment a rule that would require commercial nuclear power plant licensees to strengthen their maintenance activities in order to reduce the likelihood of failures and events caused by the lack of effective maintenance. The comment period for this proposed rule was to have expired on January 27, 1989. The Nuclear Management and Resources Council (NUMARC) has requested a sixty-day extension of the comment period. In view of the importance of the proposed rule, the amount of time that the NUMARC suggests is required in order to provide meaningful comments on behalf of its member utilities, and the desirability of developing a final rule as soon as practicable, the Commission has decided to extend the comment period for an additional thirty days. The extended comment period now expires on February 27, 1989.

DATE: The comment period has been extended and now expires February 27, 1989. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Mail written comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC.

Deliver comments to: 11155 Rockville Pike, Rockville, MD between 7:30 a.m. and 4:15 p.m. weekdays.

FOR FURTHER INFORMATION CONTACT: Moni Dey, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 492-3730.

Dated at Rockville, Maryland this 22nd day of December, 1988.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Acting Secretary for the Commission.

53 FR 52716
Published 12/29/88
Comment period expires 2/27/89

Education and Experience Requirements for Senior Reactor Operators and Supervisors at Nuclear Power Plants

See Part 55 Proposed Rule Making

54 FR 8201
Published 2/27/89
Comment period extented to 3/29/89

Education and Experience Requirements for Senior Reactor Operators and Supervisors at Nuclear Power Plants

See Part 55 Proposed Rule Making

54 FR 9229
Published 3/6/89
Comment period expires 7/5/89

10 CFR Part 50

Acceptance of Products Purchased for Use in Nuclear Power Plant Structures, Systems and Components

AGENCY: Nuclear Regulatory Commission.

ACTION: Advance notice of proposed rulemaking.

SUMMARY: The Nuclear Regulatory Commission (Commission) is proposing to develop regulations requiring enhanced acceptance procedures including, but not limited to, receipt inspection and testing of products purchased for use in nuclear power plant structures, systems and components. This Advance Notice of Proposed Rulemaking (ANPR) is intended to solicit comments on the need for additional regulatory requirements and to obtain an improved understanding of alternatives to regulatory requirements that could provide assurance that structures, systems and components procured for use in nuclear power plants will perform as expected to protect public health and safety.

DATE: The comment period expires July 5, 1989. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

ADDRESSES: Mail comments to: The Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch.

Deliver comments to: 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. Federal workdays.

Examine copies of comments received at: The NRC Public Document Room, Gelman Building, 2120 L Street NW., Washington, DC.

FOR FURTHER INFORMATION CONTACT:
Max J. Clausen, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 492-0969.

SUPPLEMENTARY INFORMATION:

Background

Appendix B to 10 CFR Part 50 of the Commission's regulations adopted in 1970 (35 FR 10498) establishes the quality assurance criteria for safety-related structures, systems and components for nuclear power plants. The purpose of the quality assurance criteria in Appendix B is to provide requirements for the design, procurement, receipt inspection and testing, construction and operation of nuclear power plant structures, systems and components. The criteria are generally structured to confirm the quality of products designed, purchased, inspected, tested and installed for use in nuclear power plant structures, systems and components. The criteria apply to all activities conducted during the design, construction and operating phases of nuclear power plants that affect the safety-related functions of structures, systems and components. Procedures and actions by licensees and their representatives conforming to these criteria are expected to detect substandard and poor quality products but may not necessarily detect counterfeit or fraudulently marketed products. Recent cases involving apparently substandard, counterfeit and fraudulently marketed products for nuclear power plant structures, systems and components have prompted the Commission to reconsider the adequacy of current regulations for detecting substandard, counterfeit and fraudulently marketed products and for assuring that such products are not used in nuclear power plant structures, systems and components.

Criteria III, IV, VII, VIII, and XV of Appendix B to 10 CFR Part 50 provide the criteria for the control of purchased structures, systems and components for nuclear power plants. Historically, licensees and their representatives have purchased products with certifications attesting to the quality of the products and have depended to varying degrees on the certifications as one basis for accepting the products. However, recent discoveries of substandard, counterfeit and fraudulently marketed products furnished to nuclear power plants by contractors and subcontractors demonstrate that current product acceptance practices, particularly those based heavily on certifications and stated catalog specifications, have not been sufficient in all cases. Additional details of apparently substandard, counterfeit, and fraudulently marketed products are contained in NRC Compliance Bulletin No. 87-02 and Supplements 1 and 2, NRC Bulletin No. 88-05 and Supplements 1 and 2, NRC Bulletin No. 88-10, NRC Information Notice No. 88-19, NRC Information Notice No. 88-35, NRC Information Notice No. 88-46 and Supplement 1, NRC Information Notice No. 88-48, and NRC Information Notice No. 88-97.[1]

In many cases, as in part discussed in the referenced bulletins and information notices, product acceptance practices have failed to detect such substandard, counterfeit or fraudulently marked products. Therefore, the Commission is considering the need for additional regulations or other methods to provide additional assurance that products purchased for use in nuclear power plant structures, systems and components satisfy requirements and specifications that are imposed to provide confidence that these items will perform as required to protect the public health and safety.

The Commissions's regulations provide two alternative approaches to assure that structures, systems and components satisfy requirements for safety-related applications. A licensee may procure products to the requirements of the applicable code or standard for the safety-related structure, system or component. Alternatively, the licensee may purchase a commercial grade product and then, using the appropriate procedures and satisfying the Commission's requirements, dedicate the commercial grade product for the safety-related application. The experiences that have been discussed in the bulletins and information notices previously referenced apply to products that were obtained using both of these approaches.

The Commission has concluded that significant engineering involvement is required during the procurement process for products used in nuclear power plants and during any testing of these products. It is the Commission's view that, in the past, inadequate engineering involvement has been a common weakness in licensees' procurement programs, particularly when commercial grade procurements were involved. It is the Commission's position that involvement of a licensee's engineering staff in the procurement process should include (1) selection of products to be used in the plant, (2) determination of the critical characteristics of the selected products that are to be verified during product acceptance, (3) determination of specific testing requirements applicable to the selected products, and (4) evaluation of test results. This involvement should be applicable to products initially procured as safety-related as well as commercial grade products procured for dedication and upgrading for use in safety-related applications. The extent of this engineering involvement will be highly dependent on the nature and use of the products involved.

The Commission is concerned about the quality of commercial grade products that are used throughout the nuclear plant, including applications in the "balance of plant" structures, sytems and components. This concern stems from a recognition that structures, sytems and components utilizing substandard products may not function as designed and may challenge safety-related systems unnecessarily or complicate the response to off-normal events. Commenters are requested to consider the issues and questions of this ANPR as they may relate to the need or desirability for either more prescriptive regulations or, alternatively, a performance-based requirement. Comments are also requested on the desirability of any such requirement for safety-related applications, as well as for applications throughout the plant.

A broad spectrum of issues needs to be considered before the Commission decides on the scope and content of any proposed new regulatory requirements addressing the concerns raised by the experience of licensees placing essentially complete reliance on certificates, such as Certificates of Compliance, and the evidence that some contractors are misrepresenting products. This experience is discussed in the referenced bulletins and information notices. The following questions are posed to raise the issues that the Commission has identified and are not to be considered complete nor are they intended to bound the scope of public comment on this ANPR. The questions are structured in two categories: (1) Products Procured for Use in Safety-Related Structure, System and Component Applications, and (2) Dedication of Commercial Grade Products for Use in Safety-Related Structure, System and Component Applications.

Public comments are invited on each of these questions. The comment resolution process will be improved if each comment is identified to the question to which it responds. Commenters may submit, in addition to the original paper copy, a copy of the letter in an electronic format on IBM PC DOS compatible 3.5 or 5.25 inch double sided double density (DS/DD) diskettes. Data files should be provided in ASCII code or, if formatted text is required, data files should be provided in IBM Revisable-Form Text Document Content Architecture (RFT/DCA) format.

[1] These documents are available for inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street, NW., Washington, DC.

1. Products Procured for Use in Safety-Related Structure, System and Component Applications.

The questions in this section are categorized in four subsections: General, Metallic Products, Nonmetallic Products, and Components.

1.1 General

1.1.1a In view of the problems that have been detected with substandard, counterfeit, or fraudulently marketed products, do the Commission's current regulations provide adequate criteria for ensuring the acceptability of purchased products?

1.1.1b If the current regulations are considered to provide adequate criteria, how should they be applied to ensure that substandard, counterfeit, and fraudulently marketed products are detected and precluded from use in nuclear power plants?

1.1.1c If the current regulations do not provide adequate criteria, should the Commission establish specific requirements or performance-based requirements to ensure that products purchased for use in nuclear power plant structures, systems and components satisfy the operational requirements necessary to protect public health safety?

1.1.2a What traceability requirements should be imposed for all products to be used in safety-related structures, systems and components?

1.1.2b Should material traceability through all intermediary contractors, subcontractors and processors be required?

1.1.2c Should all critical characteristics, for example, materials, operations, functions, etc., be traceable?

1.1.2d Should there be any exceptions to the traceability requirements?

1.1.2e What should the identification requirements be for traceability, for example, uniquely marking each part whenever possible, bagging, records, etc.?

1.1.3 Should product acceptances be restricted to inspections and tests or should product acceptances include, on a sample basis, destructive inspections and tests to verify chemical and physical characteristics?

1.1.4 What types of inspections and tests (appropriate for the various types of products) should be required?

1.1.5 Should licensees, contractors and subcontractors be encouraged to perform joint testing?

1.1.6 If destructive inspections and tests are determined necessary, what should be the sampling basis (per vendor, per purchase order, per shipment, per lot, per container, etc.)?

1.1.7 Should sample plan testing be permitted for testing or should such testing be on a 100 percent basis?

1.1.8 What sort of statistical sampling during product inspection is adequate to provide confidence that the product has the requisite assurance of quality?

1.1.9 What criteria should be used for allowing sample plan testing during product acceptance?

1.1.10 Should the shelf life of appropriate types of structures, systems and components be inspected and verified as acceptable during product acceptances?

1.1.11 To what extent will an effective vendor audit program and maintenance of a qualified vendor list reduce the likelihood of questionable products being used in nuclear power plants?

1.1.12 What are the essential elements, for example, team composition, depth of audits, and approach, that must be included in an effective vendor audit program?

1.1.13 What reinspection or reaudit frequency is appropriate to maintain confidence in those vendors on a qualified vendor list?

1.1.14 How do licensees ensure that Code Certificate holders and "N" stamp vendors are current?

1.1.15 Is there an auditable method to demonstrate that licensees actually purchased the product from a qualified vendor, for example, the holder of an ASME Code stamp holder certificate?

1.1.16a Should negative inspection, testing and audit results be shared with other parties?

1.1.16b Is a Federal requirement necessary to permit this sharing of information?

1.1.16c Should procurement contracts be required to include a provision for public release of the results of audits of the vendor?

1.1.16d Are there restraint of trade, antitrust concerns or liabilities associated with these actions?

1.1.17 Should licensees, contractors and subcontractors be encouraged to make joint procurements and to share inspection/audit results of joint procurements to enhance the effectiveness of inspections/audits?

1.1.18 If joint procurements and inspections/audits are encouraged, should controls be imposed and, if so, what and how should these controls be imposed?

1.1.19 What audit and testing documentation should be required to provide traceability and give confidence to all participants in joint product acceptances?

1.1.20a Should the NRC establish and publish a list of approved vendors for various products?

1.1.20b If a list of approved vendors is established, how should vendors be selected?

1.1.20c If a list of approved vendors is established, who should be responsible for maintaining this list?

1.1.20d Should licensees be restricted to making procurements from this list of approved vendors?

1.1.21 Should the use of a certificate, such as a Certificate of Conformance, in the procurement process be prohibited or, if allowed, be restricted to issue by the original equipment manufacturer for items that have remained under his direct control?

1.1.22 Should the furnishing of the original manufacturer's certificate, such as a Certified Material Test Report, be made mandatory for procurement of materials from intermediate vendors?

1.1.23 Should the transcribing of an original manufacturer's test data by intermediate vendors onto the vendor's certification, for example, Certified Material Test Report, be forbidden?

1.1.24 To what extent should licensees or their representatives be required to inspect the implementation of contractor product acceptance programs?

1.1.25 Should licensees be required to audit implementation of 10 CFR Part 21 by suppliers and vendors?

1.1.26 In addition to the requirements of 10 CFR Part 21, should licensees be required to notify manufacturers, suppliers and vendors of licensee-identified problems with vendor-provided nonconforming products or programs?

1.1.27 Should licensee participation in a national data system for reporting equipment/component failures by manufacturer and application be required?

1.1.28 Is there specific data that should be included in a national data system that would significantly enhance its usefulness in establishing equipment performance history?

1.1.29 What are the implications of any new Commission requirements on the Commission's endorsement of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code in 10 CFR 50.55a?

1.1.30 What is the best way to coordinate any new requirements with the ASME Boiler and Pressure Vessel Code?

1.1.31 Should the new requirements that relate to areas covered by the ASME Boiler and Pressure Vessel Code (e.g., SA material specifications) be handled through the code committee system?

1.1.32 To what extent should items 1.1.1a through 1.1.31 be required for other than safety-related components?

1.2 Metallic Products (e.g., fasteners, piping, pipe fittings, weld rod, castings, forgings, bar stock, plate material, stampings, wire, cable, etc.).

1.2.1a Should chemical analyses of the products be required as part of product acceptances?

1.2.1b Should these analyses of the products be performed by destructive or by nondestructive means?

1.2.2a Should tests of mechanical properties (e.g., hardness, tensile strength, impact, etc.) of the products be required as part of product acceptances?

1.2.2b Should tests of mechanical properties of the products be performed by destructive or by nondestructive means?

1.2.3 When destructive tests are required, are test coupons (when applicable) an acceptable source of test materials for the tests of chemical and mechanical properties or should material samples be removed from actual products?

1.3 Nonmetallic Products (e.g., lubricants, tape, elastomers, seals, paints, filters, etc.)

1.3.1a Should chemical analyses be required for lubricants, tape, elastomers, etc., during product acceptances?

1.3.1b Should these analyses be performed by destructive or by nondestructive means?

1.3.2 Should physical property tests (e.g., viscosity for lubricants, hardness for elastomers, efficiency for filters, etc.) be required during product acceptances?

1.4 Components (e.g., pumps, valves, circuit breakers, controllers and electronic parts/assemblies and their replacement parts)

1.4.1 Should components be subjected to functional tests during product acceptance?

1.4.2a Should components be disassembled, if necessary during product acceptance, to verify dimensional characteristics?

1.4.2b If the components are not disassembled, what methods should be utilized to verify critical characteristics?

1.4.3a Should the chemical and physical properties of component materials be analyzed during product acceptance inspections?

1.4.3b If the chemical and physical properties of component materials are to be analyzed during product acceptance inspections, what means should be utilized?

2. Dedication of Commercial Grade Products for Use in Safety-Related Structure, System and component Applications.

The questions in this section are categorized in five subsections: General, Metallic Products, Nonmetallic Products, Components, and Others Questions.

2.1 General

2.1.1 Should the Commission establish specific requirements or performance-based requirements to ensure that commercial grade products being dedicated for use in safety-related nuclear power plant structures, systems and components satisfy the operational requirements necessary to protect public health and safety?

2.1.2 Should NRC regulations be revised to endorse and incorporate by reference, the industry codes, standards, or guidance documents for dedication programs of commercial grade products for use in safety-related structure, system and component applications?

2.1.3a What should the traceability requirements be for all commercial products being upgraded for use in safety-related structures, systems and components?

2.1.3b If upgrading traceability provisions are necessary, what should these provisions include?

2.1.3c Should material traceability through all intermediary contracts, subcontractors and processors be required?

2.1.3d If item traceability is necessary, should there be any provisions for upgrading products whose traceability cannot be established?

2.1.3e Should the upgrading traceability provisions be any different if the products are heat/lot identified or not?

2.1.3f What should the identification requirements be for traceability, for example, marking, bagging and records?

2.1.4 How should products intended for use in applications in which products are normally required to meet specific standards be inspected to verify that all critical characteristics are satisfied?

2.1.5 Should the shelf life of appropriate types of products be inspected and verified as acceptance as part of the upgrade inspection process?

2.1.6 What types of shelf life controls should be imposed on products that are being upgraded for use in safety-related structures, systems and components?

2.1.7 Should all upgrade inspections be restricted to inspections and tests or should they include, on a sample basis, destructive inspections and tests to verify chemical and physical characteristics?

2.1.8 What types of inspections and tests (appropriate for the various types of products) should be required?

2.1.9 How should inspections verify all critical characteristics (for example, chemistry, physical properties, dimensions, special processes, etc.)?

2.1.10a If destructive inspections and tests are determined to be necessary, how should samples be selected if products are heat/lot identified?

2.1.10b How should samples be selected if products are not heat/lot identified?

2.1.11 Should sample plan testing be permitted for nondestructive testing or should such testing be on a 100 percent basis?

2.1.12 What criteria are appropriate for allowing sample plan testing during upgrade inspections?

2.2 Metallic Products

2.2.1a Are chemical analyses of the products appropriate as part of upgrade inspections?

2.2.1b Should these analyses be performed by destructive or by nondestructive means?

2.2.2a Are tests of mechanical properties (e.g., hardness, tensile strength, impact etc.) appropriate as part of upgrade inspections?

2.2.2b Should these tests be performed by destructive or by nondestructive means?

2.2.3 If the product is heat/lot traceable, is sample inspection (destructive and nondestructive) adequate for confirmation of critical characteristics?

2.2.4 If the product is not heat/lot traceable, is it necessary to either sample or 100 percent test, for example, hardness, to establish uniformity and then destructively analyze them (for example, chemical analyses, tensile tested, impact tested, etc.) to determine acceptability?

2.2.5 Should requirements in addition to those included in industry standards (e.g., additional samples, etc.) be required?

2.2.6 When destructive tests are required, are test coupons (when available) an acceptable source of test materials for chemical and mechanical properties tests or should material samples be removed from actual products?

2.3 Nonmetallic Products

2.3.1a Are chemical analyses necessary to establish critical characteristics for lubricants, tape, elastomers, etc., proposed for upgrading for use in safety-related systems?

2.3.1b Should these analyses be performed by destructive or by nondestructive means?

2.3.2 Are physical property tests (e.g., viscosity for lubricants, hardness

for elastomers, efficiency for filters, etc.) necessary for upgrading these products?

2.3.3 May critical characteristics be inspected by samples or is 100 percent inspection necessary to verify these characteristics?

2.4 Components

2.4.1 Must each critical characteristic be inspected before the component is acceptable for use in safety-related systems?

2.4.2 How should the chemical and physical properties of component materials be analyzed during upgrade inspections?

2.4.3 If critical characteristics cannot be inspected on each component piece, should it be acceptable to establish heat/lot traceability, establish uniformity of lot by sample inspection and thereby accept the lot?

2.4.4a Must components be 100 percent functionally tested or may they be subjected to functional tests on a sampling basis?

2.4.4b Inspected by sample, what is the basis for performing only sample inspection?

2.4.5a Should disassembly of components be required to verify critical characteristics?

2.4.5b May verification of critical characteristics be done on a sampling basis or are 100 percent inspections necessary?

2.4.5c What is the basis for performing only sample inspections?

2.4.5d If components are not disassembled to verify dimensions, what methods can be utilized to verify dimensions?

2.5 Other Questions

2.5.1a Are there any other agency/organization standards or programs that should be adopted for use in upgrading commercial grade products for use in safety-related systems?

2.5.1b Should these standards or programs be endorsed by NRC regulations?

2.5.2 Are there other alternatives that could provide the necessary assurances?

2.5.3 To what extent should any existing controls or any additional controls being contemplated in the ANPR be extended to nonsafety-related applications in "balance of plant" structures, systems and components?

List of Subjects in 10 CFR Part 50

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

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Dated at Rockville, Maryland, this 28th day of February, 1989.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

36 FR 3255

Criterion 4—Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

52 FR 41288

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

36 FR 3255

36 FR 12733

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

36 FR 12733

Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

36 FR 3255

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

36 FR 3255

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

36 FR 12733

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

36 FR 12733

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18—Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

36 FR 3255

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall for high functional reliability

and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity [illegible] design and principles of operation [illegible] to the extent practical to [illegible] of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat re-

moval system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close

to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall 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.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling systems shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

APPENDIX B—QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation.

Nuclear powerplants and fuel reprocessing plants* include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

As used in this appendix, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

I. ORGANIZATION

The applicant shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but shall retain responsibility therefor. The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have this required authority and organizational freedom, irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this Appendix are being performed shall have direct access to such levels of management as may be necessary to perform this function.

II. QUALITY ASSURANCE PROGRAM

The applicant shall establish at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety. Activities affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, such as adequate cleanness, and assurance that all prerequisites for the given activity have been satisfied. The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test. The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

III. DESIGN CONTROL

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participat-

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*While the term "applicant" is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear power plant or a fuel reprocessing plant. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

*Amended 36 FR 18301.

ing design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualification testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

35 FR 10498

IV. PROCUREMENT DOCUMENT CONTROL

Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.

V. INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

VI. DOCUMENT CONTROL

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

VII. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear powerplant or fuel reprocessing plant* site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear powerplant or fuel reprocessing plant* site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment. The effectiveness of the control of quality by contractors and subcontractors shall be assessed by the applicant or designee at intervals consistent with the importance, complexity, and quantity of the product or services.

VIII. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

IX. CONTROL OF SPECIAL PROCESSES

Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

X. INSPECTION

A program for inspection of activities affecting quality shall be established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Such inspection shall be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed shall be performed for each work operation where necessary to assure quality. If inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel shall be provided. Both inspection and process monitoring shall be provided when control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the applicant's designated representative and beyond which work shall not proceed without the consent of its designated representative are required, the specific hold points shall be indicated in appropriate documents.

XI. TEST CONTROL

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear powerplant or fuel reprocessing plant* operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

XII. CONTROL OF MEASURING AND TEST EQUIPMENT

Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

XIII. HANDLING, STORAGE, AND SHIPPING

Measures shall be established to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.

XIV. INSPECTION, TEST, AND OPERATING STATUS

Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant or fuel reprocessing plant.* These measures shall provide for the identification of items which have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of such inspections and tests. Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant or fuel reprocessing plant,* such as by tagging valves and switches, to prevent inadvertent operation.

35 FR 10498

XV. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

XVI. CORRECTIVE ACTION

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of significant conditions adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

XVII. QUALITY ASSURANCE RECORDS

Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following: Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment. Inspection and test records shall, as a mini-

*Amended 36 FR 18301

mum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiencies noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility.

XVIII. AUDITS

A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Followup action, including reaudit of deficient areas, shall be taken where indicated.

35 FR 10430

Appendix C—A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Facility Construction Permits

General Information

This appendix is intended to apprise applicants for licenses to construct production or utilization facilities of the types described in § 50.21(b) or § 50.22, or testing facilities, of the general kinds of financial data and other related information that will demonstrate the financial qualification of the applicant to carry out the activities for which the permit is sought. The kind and depth of information described in this guide is not intended to be a rigid and absolute requirement. In some instances, additional pertinent material may be needed. In any case, the applicant should include information other than that specified, if such information is pertinent to establishing the applicant's financial ability to construct the proposed facility.

It is important to observe also that both § 50.33(f) and this appendix distinguish between applicants which are established organizations and those which are newly-formed entities organized primarily for the purpose of engaging in the activity for which the permit is sought. Those in the former category will normally have a history of operating experience and be able to submit financial statements reflecting the financial results of past operations. With respect, however, to the applicant which is a newly formed company established primarily for the purpose of carrying out the licensed activity, with little or no prior operating history, somewhat more detailed data and supporting documentation will generally be necessary. For this reason, the appendix describes separately the scope of information to be included in applications by each of these two classes of applicants.

In determining an applicant's financial qualification, the Commission will require the minimum amount of information necessary for that purpose. No special forms are prescribed for submitting the information. In many cases, the financial information usually contained in current annual financial reports, including summary data of prior years, will be sufficient for the Commission's needs. The Commission reserves the right, however, to require additional financial information at the construction permit stage, particularly in cases in which the proposed power generating facility will be commonly owned by two or more existing companies or in which financing depends upon long-term arrangements for sharing of the power from the facility by two or more electrical generating companies.

Applicants are encouraged to consult with the Commission with respect to any questions they may have relating to the requirements of the Commission's regulations or the information set forth in this appendix.

I. Applicants Which Are Established Organizations

A. Applications for construction permits

1. *Estimate of construction costs.* For electric utilities, each applicant's estimate of the total cost of the proposed facility should be broken down as follows and be accompanied by a statement describing the bases from which the estimate is derived:

(a) Total nuclear production plant costs	$
(b) Transmission, distribution, and general plant costs	$
(c) Nuclear fuel inventory cost for first core [1]	$
Total estimated cost	$

[1] Section 2.790 of 10 CFR Part 2 and § 9.5 of 10 CFR Part 9 indicate the circumstances under which information submitted by applicants may be withheld from public disclosure.

49 FR 35747

If the fuel is to be acquired by lease or other arrangement than purchase, the application should so state. The items to be included in these categories should be the same as those defined in the applicable electric plant and nuclear fuel inventory accounts prescribed by the Federal Energy Regulatory Commission or an explanation given as to any departure therefrom.

Since the composition of construction cost estimates for production and utilization facilities other than nuclear power reactors will vary according to the type of facility, no particular format is suggested for submitting such estimates. The estimate should, however, be itemized by categories of cost in sufficient detail to permit an evaluation of its reasonableness.

50 FR 18652

2. *Source of construction funds.* The application should include a brief statement of the applicant's general financial plan for financing the cost of the facility, identifying the source or sources upon which the applicant relies for the necessary construction funds, e.g., internal sources such as undistributed earnings and depreciation accruals, or external sources such as borrowings.

49 FR 35747

3. *Applicant's financial statements.* The application should also include the applicant's latest published annual financial report, together with any current interim financial statements that are pertinent. If an annual financial report is not published, the balance sheet and operating statement covering the latest complete accounting year together with all pertinent notes thereto and certification by a public accountant should be furnished.

II. Applicants Which Are Newly Formed Entities

A. Applications for construction permits

1. *Estimate of construction costs.* The information that will normally be required of applicants which are newly formed entities will not differ in scope from that required of established organizations. Accordingly, applicants should submit estimates as described above for established organizations.

2. *Source of construction funds.* The application should specifically identify the source or sources upon which the applicant relies for the funds necessary to pay the cost of constructing the facility, and the amount to be obtained from each. With respect to each source, the application should describe in detail the applicant's legal and financial relationships with its stockholders, corporate affiliates, or others (such as financial institutions) upon which the applicant is relying for financial assistance. If the sources of funds relied upon include parent companies or other corporate affiliates, information to support the financial capability of each such company or affiliate to meet its commitments to the applicant should be set forth in the application. This information should be of the same kind and scope as would be required if the parent companies or affiliates were in fact the applicant. Ordinarily, it will be necessary that copies of agreements or contracts among the companies be submitted.

49 FR 35747

As noted earlier in this appendix, an applicant which is a newly formed entity will normally not be in a position to submit the usual types of balance sheets and income statements reflecting the results of prior operations. The applicant should, however, include in its application a statement of its assets, liabilities, and capital structure as of the date of the application.

III. Annual Financial Statement

Each holder of a construction permit for a production or utilization facility of a type described in § 50.21(b) or § 50.22, or a testing facility is required by § 50.71(b) to file its annual financial report with the Commission at the time of issuance thereof. This requirement does not apply to licensees or holders of construction permits for medical and research reactors.

50 FR 18652

IV. Additional Information

The Commission may, from time to time, request the applicant, whether an established organization or newly formed entity, to submit additional or more detailed information respecting its financial arrangements and status of funds if such information is deemed necessary to enable the Commission to determine an applicant's financial qualifications for the license.

Appendix D (Removed 39 FR 26279)

Appendix E—Emergency Planning and Preparedness for Production and Utilization Facilities

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I. Introduction
II. The Preliminary Safety Analysis Report
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V. Implementing Procedures

45 FR 55402

I. Introduction

Each applicant for a construction permit is required by § 50.34(a) to include in the preliminary safety analysis report a discussion of preliminary plans for coping with emergencies. Each applicant for an operating license is required by § 50.34(b) to include in the final safety analysis report plans for coping with emergencies.

This appendix establishes minimum requirements for emergency plans for use in attaining an acceptable state of emergency preparedness. These plans shall be described generally in the preliminary safety analysis report and submitted as part of the final safety analysis report.

The potential radiological hazards to the public associated with the operation of research and test reactors and fuel facilities licensed under 10 CFR Parts 50 and 70 involve considerations different than those associated with nuclear power reactors. Consequently, the size of Emergency Planning Zones [1] (EPZs) for facilities other than power reactors and the degree to which compliance with the requirements of this section and sections II, III, IV, and V as necessary will be determined on a case-by-case basis.[2]

Notwithstanding the above paragraphs, in the case of an operating license authorizing only fuel loading and/or low power operations up to 5% of rated power, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and the capability to implement State and local offsite emergency plans, as defined in this Appendix, are required prior to the issuance of such a license.

[1] EPZs for power reactors are discussed in NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978. The size of the EPZs for a nuclear power plant shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. Generally, the plume exposure pathway EPZ for nuclear power plants with an authorized power level greater than 250 MW thermal shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius.

[2] Regulatory Guide 2.6 will be used as guidance for the acceptability of research and test reactor emergency response plans.

II. The Preliminary Safety Analysis Report

The Preliminary Safety Analysis Report shall contain sufficient information to ensure the compatibility of proposed emergency plans for both onsite areas and the EPZs, with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, land use, and local jurisdictional boundaries for the EPZs in the case of nuclear power reactors as well as the means by which the standards of § 50.47(b) will be met.

As a minimum, the following items shall be described:

A. Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

B. Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

C. Protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which those measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency.

(D) Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

E. Provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

F. Provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

G. A preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. A nuclear power plant applicant shall perform a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or taking of protective actions.

H. A preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the onsite technical support center and of the near-site emergency operations facility in assessing information, recommending protective action, and disseminating information to the public.

III. The Final Safety Analysis Report

The Final Safety Analysis Report shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee.

The plans submitted must include a description of the elements set out in Section IV for the Emergency Planning Zones (EPZs) [1] to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.

IV. Content of Emergency Plans

The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiation emergencies, assessment action, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license shall contain information needed to demonstrate compliance with the standards described in Section 50.47(b), and they will be evaluated against those standards. The nuclear power reactor operating license applicant shall also provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.

A. Organization

The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. Specifically, the following shall be included:

1. A description of the normal plant operating organization.

2. A description of the onsite emergency response organization with a detailed discussion of:

a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency;

b. Plant staff emergency assignments;

c. Authorities, responsibilities, and duties on an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.

3. A description, by position and function to be performed, of the licensee's headquarters personnel who will be relied to the plant site to augment the onsite emergency organization.

4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.

5. Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications (such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.

6. A description of the local offsite services to be provided in support of the licensee's emergency organization.

7. Identification of, and assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies.

8. Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.

B. Assessment Actions

The means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.

C. Activation of Emergency Organization

The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG–0654/FEMA–REP–1.

D. Notification Procedures

1. Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs.[1]

2. Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.

3. A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By February 1, 1982, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The four-month period in 10 CFR 50.54(s)(2) for the correction of emergency plan deficiencies shall not apply to the initial installation of this public notification system that is required by February 1, 1982. *The four-month period will apply to correction of deficiencies identified during the initial installation and testing of the prompt public notification systems as well as those deficiencies discovered thereafter.* The design objective of the prompt public notification system shall be to have the capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this notification capability will range from immediate notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the State and local governmental officials to make a judgment whether or not to activate the public notification system. Where there is a decision to activate the notification system, the State and local officials will determine whether to activate the entire notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public notification system shall remain with the appropriate governmental authorities.

E. Emergency Facilities and Equipment

Adequate provisions shall be made and described for emergency facilities and equipment, including:

1. Equipment at the site for personnel monitoring;

2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

3. Facilities and supplies at the site for decontamination of onsite individuals;

4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on site;

6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;

8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

9. At least one onsite and one offsite communications system; each system shall have a backup power source.

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

a. Provision for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly.

F. Training

The program to provide for (1) the training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (2) the participation in the training and drills by other persons whose assistance may be needed in the event of a radiation emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:

a. Directors and/or coordinators of the plant emergency organization;

b. Personnel responsible for accident assessment, including control room shift personnel;

c. Radiological monitoring teams;

d. Fire control teams (fire brigades);

e. Repair and damage control teams;

f. First aid and rescue teams;

g. Medical support personnel;

h. Licensee's headquarters support personnel;

i. Security personnel.

In addition, a radiological orientation training program shall be made available to

local services personnel; e.g., local emergency services/Civil Defense, local law enforcement personnel; local news media persons.

The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, test the public notification system, and ensure that emergency organization personnel are familiar with their duties.[3]

1. A full participation[4] exercise which tests as much of the licensee, State and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located for which the first operating license for that site is issued after July 13, 1982. This exercise shall be conducted within two years before the issuance of the first operating license for full power (one authorizing operation above 5% of rated power) of the first reactor and shall include participation by each State and local government within the plume exposure pathway EPZ and each State within the ingestion exposure pathway EPZ. If the full participation exercise is conducted more than one year prior to issuance of an operating license for full power, an exercise which tests the licensee's onsite emergency plans shall be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.

2. Each licensee at each site shall annually exercise its emergency plan.

3. Each licensee at each site shall exercise with offsite authorities such that the State and local government emergency plans for each operating reactor site are exercised biennially, with full or partial participation [5] by States and local governments within the plume exposure pathway EPZ. State and local governments that have fully participated in a joint exercise since October 1, 1982, are eligible to fully participate in emergency preparedness exercises on a biennial frequency. The level of participation shall be as follows:

(a) A State shall at least partially participate in each offsite exercise at each site.

(b) A State shall fully participate in at least one offsite exercise every 2 years.

(c) At least once every 7 years, all States within the plume exposure pathway EPZ for a given site must fully participate in an offsite exercise for that site. This exercise must also involve full participation by local governments within the plume exposure pathway EPZ.

(d) Partial participation by a local government during an offsite exercise for a site is acceptable only when the local government is fully participating in a biennial exercise at another site.

(e) Each State within any ingestion exposure pathway EPZ shall exercise its plans and preparedness related to ingestion exposure pathway measures at least once every 5 years.

(f) Licensees shall enable any State or local government located within the plume exposure pathway EPZ to participate in annual exercises when requested by such State or local government.

4. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC, in consultation with FEMA, cannot find reasonable assurance that adequate protective measures can be taken in the event of a radiological emergency. The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.

5. All training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified shall be corrected.

6. The participation of state and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to 10 CFR 50.47(c)(1). In such cases, an exercise shall be held with the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

H. Recovery

Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.

V. Implementing Procedures

No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in § 50.4. Licensees who are authorized to operate a nuclear power facility shall submit any changes to the emergency plan or procedures to the Commission, as specified in § 50.4, within 30 days of such changes.

[3] Use of site specific simulators or computers is acceptable for any exercise.

[4] "Full participation" when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite local and State authorities and licensee personnel physically and actively take part in testing their integrated capability to adequately assess and respond to an accident at a commercial nuclear power plant. "Full participation" includes testing the major observable portions of the onsite and offsite emergency plans and mobilization of State, local and licensee personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.

[5] "Partial participation" when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite authorities shall actively take part in the exercise sufficient to test direction and control functions; i.e., (a) protective action decision making related to emergency action levels, and (b) communication capabilities among affected State and local authorities and the licensee.

APPENDIX F

POLICY RELATING TO THE SITING OF FUEL REPROCESSING PLANTS AND RELATED WASTE MANAGEMENT FACILITIES

1. Public health and safety considerations relating to licensed fuel reprocessing plants do not require that such facilities be located on land owned and controlled by the Federal Government. Such plants, including the facilities for the temporary storage of high-level radioactive wastes, may be located on privately owned property.

2. A fuel reprocessing plant's inventory of high-level liquid radioactive wastes will be limited to that produced in the prior 5 years. (For the purpose of this statement of policy, "high-level liquid radioactive wastes" means those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels.) High-level liquid radioactive wastes shall be converted to a dry solid as required to comply with this inventory limitation, and placed in a sealed container prior to transfer to a Federal repository in a shipping cask meeting the requirements of 10 CFR Part 71. The dry solid shall be chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository. All of these high-level radioactive wastes shall be transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel. Upon receipt, the Federal repository will assume permanent custody of these radioactive waste materials although industry will pay the Federal Government a charge which together with interest on unexpended balances will be designed to defray all costs of disposal and perpetual surveillance. The Department of Energy will take title to the radioactive waste material upon transfer to a Federal repository. Before retirement of the reprocessing plant from operational status and before termination of licensing pursuant to § 50.82, transfer of all such wastes to a Federal repository shall be completed. Federal repositories, which will be limited in number, will be designated later by the Commission.

35 FR 17530

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

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

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

36 FR 5481

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

Amended 45FR14199

Appendix G—Fracture Toughness Requirements

Table of Contents

I. Introduction and Scope
II. Definitions
III. Fracture Toughness Tests
IV. Fracture Toughness Requirements
V. Inservice Requirements—Reactor Vessel Beltline Materials

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this Appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in § 50.55a of this part.

48 FR 24008

The ASME Boiler and Pressure Vessel Code has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register**. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 2120 L Street NW., Washington, D.C.

53 FR 43419

The requirements of this appendix apply to the following materials:

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

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

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

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

II. Definitions

A. "Ferritic material" means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. "System hydrostatic tests" means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. "Specified minimum yield strength" means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under § 50.55a of this part.

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

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

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

48 FR 24008

III. Fracture Toughness Tests

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

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

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

IV. Fracture Toughness Requirements

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

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

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

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

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

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

B. Reactor vessels for which the predicted

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

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

V. Inservice Requirements—Reactor Vessel Beltline Material

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

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this appendix are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions are to be made for the radiation conditions at the critical location on the crack front of the assumed flaw.[1] The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials must be used to verify that the fracture toughness requirements are satisfied.

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

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

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

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

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

E. The proposed programs for satisfying the requirements of sections V.C. and V.D. of this appendix must be submitted, as specified in § 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of section V.B. of this appendix.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

Table of Contents

I. Introduction
II. Surveillance Program Criteria
III. Report of Test Results

I. Introduction

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

ASTM E 185-73, -79 and -82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of ASTM E 185-73, -79, and -82, may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103. Copies will be available for inspection at the Commission's Public Document Room, 2120 L Street NW., Washington, D.C.

II. Surveillance Program Criteria

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

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

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

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

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

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

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

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

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

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

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

III. Report of Test Results

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

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

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

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

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

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

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

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

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

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

2. Notwithstanding the guidance of paragraph B.1:

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

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

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

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis.

The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25-30, reproduced in the Annex to this Appendix I.

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

2. The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

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

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

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

40 FR 19435

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

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

40 FR 58847

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

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

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

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

42 FR 20138

A. If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water-cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall: [1]

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

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

51 FR 40353

3. Report these actions as specified in § 50.4, within 30 days from the end of the quarter during which the release occurred.

40 FR 19429

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

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

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

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

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

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

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

40 FR 58847

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

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

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2) GUIDES ON DESIGN OBJECTIVES FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

40 FR 45816

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

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

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

3. Notwithstanding the guidance in paragraph A.2. for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures [1] to reduce the possible sources of radioactive material in liquid effluent releases and the calculated quantity exceeds the quantity set forth in paragraph A.2. the requirements for design objectives for radioactive material in liquid effluents may be deemed to have been met provided:

51 FR 40330

a. The applicant submits, as specified in § 50.4, an evaluation of the potential for effects from long-term buildup on the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

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

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

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

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

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

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

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

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

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

b. Design objectives based on a higher quantity of radioactive material above background in gaseous effluents to be released to the atmosphere than the quantity specified in paragraphs B.1 and B.2 may be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in annual doses to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin.

[40 F.R. 58847]

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

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

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

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

[40 F.R. 58847]

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

APPENDIX J

PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

I. Introduction.
II. Explanation of terms.
III. Leakage test requirements.
A. Type A test.
B. Type B test.
C. Type C test.
D. Periodic retest schedule.
IV. Special test requirements.
A. Containment modifications.
B. Multiple leakage-barrier containments.
V. Inspection and reporting of tests.
A. Containment inspection.
B. Report of test results.

I. Introduction

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

[38 F.R. 4385]

II. Explanation of Terms

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

III. Leakage Testing Requirements

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

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

A. *Type A test*—1. *Pretest requirements.* (a) Containment inspection in accordance with V.A. shall be performed as a prerequisite to the performance of Type A tests. During the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested in as close to the "as is" condition as practical. During the period between the

completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments shall be made to components whose leakage exceeds that specified in the technical specification as soon as practical after identification. If during a Type A test, including the supplemental test specified in III.A.5.(b), potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test or which result in the Type A test not meeting the acceptance criteria III.A.6.(b) or III.A.5.(b), the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and or adjustments to equipment shall be made and a Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from the local leak and Type A tests shall be included in the report submitted to the Commission as specified in V.B.

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

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

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

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

3. *Test Methods.* (a) All Type A tests shall be conducted in accordance with the provisions of the American National Standard N45.4–1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors", March 16, 1972. In addition to the Total time and Point-to-Point methods described in that standard, the Mass Point Method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in the American National Standard ANSI/ANS 56.8–1987, "Containment System Leakage Testing Requirements," [Allowed] 20, 1987.[1]

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

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

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

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

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

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

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

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

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

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

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

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

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

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

B. *Type B tests.*

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

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

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

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

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

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

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

C. *Type C tests.*

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

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

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

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

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

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

D. *Periodic retest schedule*—1. *Type A test.* (a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant in-service inspections.[3]

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

[1] ANSI N45.4–1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972. Incorporation of ANSI N45.4–1972 by reference was approved by the Director of the Federal Register on October 31, 1972. Copies of this standard may be purchased from ANSI/ANS 56.8–1987, "Containment System Leakage Testing Requirements," [Allowed] 20, 1987, may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60525.

[2] A copy of each of these standards is available for inspection at the Commission's Public Document Room at 2120 L Street, NW., Washington, DC.

[3] Amended 38 FR 5907

[3] Such in-service inspections are required

38 FR 4385

38 FR 4385

38 FR 4385

53 FR 45890

November 30, 1988 50-70

integrity. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests, except for tests of air locks, may, notwithstanding the test schedule specified under III.D.1., be performed every other reactor shutdown for refueling but in no case at intervals greater than 3 years.

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

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

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

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

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

IV. Special Testing Requirements

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

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

V. Inspection and Reporting of Tests

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

B. *Report of test results.* 1. The preoperational and periodic tests must be the subject of a summary technical report submitted to the Commission, as specified in § 50.4, approximately three months after the conduct of each test. The report must be titled "Reactor Containment Building Integrated Leak Rate Test."

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

3. For each periodic test, leakage test results from Type A, B, and C tests shall be reported. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test. Leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria of III.A.5(b), III.B.2, and III.C.3, respectively, shall be reported in a separate accompanying summary report that includes an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

* Amended 38 FR 5997.

† Amended 41 FR 16445.

Appendix K—ECCS Evaluation Models

I. Required and Acceptable Features of Evaluation Models.

II. Required Documentation.

I. Required and Acceptable Features of the Evaluation Models

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

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

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

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

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

5. *Metal-Water Reaction Rate.* The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L. C., "Studies of Metal-Water Reactions at High Temperatures. III. Experimental and Theoretical Studies of the Zirconium-Water Reac-

tion," ANL-6548, page 7, May 1962) The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall also be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

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

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

B. SWELLING AND RUPTURE OF THE CLADDING AND FUEL ROD THERMAL PARAMETERS

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

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

C. BLOWDOWN PHENOMENA

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

b. *Discharge Model.* For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," *Journal of Heat Transfer, Trans American Society of Mechanical Engineers*, 87, No. 1, February, 1955). The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperature calculated by this variation has been achieved.

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

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

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

3. *Momentum Equation.* The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

4. *Critical Heat Flux.* a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *W-3.* L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," *Journal of Nuclear Energy*, Vol. 21, 241-248, 1967.

(2) *B&W-2.* J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," *Two-Phase Flow and Heat Transfer in Rod Bundles*, ASME, New York, 1969.

(3) *Hench-Levy.* J. M. Healzer, J. E. Hench, E. Janssen, S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, GE Company Private report, July 1966.

(4) *Macbeth.* R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," *Proceedings of the Institute of Mechanical Engineers*, 1965-1966.

(5) *Barnett.* P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.

(6) *Hughes.* E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412, Idaho Nuclear Corporation, July 1970.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *GE transient CHF.* B. C. Slifer, J. E. Hench, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, General Electric Company, Equation C-32, April 1971.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown, even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

5. *Post-CHF Heat Transfer Correlations.* a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

39 FR 1003

53 FR 35996

b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969) and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," USNRC Docket RM-50-1, page 25-1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J.B. McDonough, W. Milich, E.C. King, "An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube," Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197–208, (1961) is suitable for use between nucleate and film boiling. Use of all these correlations is restricted as follows:

39 FR 1001

(1) The Groeneveld correlation shall not be used in the region near its low-pressure singularity.

(2) The first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300 °F.

(3) Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 °F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

53 FR 35996

c. Evaluation models approved after October 17, 1988, which make use of the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities," MIT Report Number 9079-28, Cambridge, Massachusetts, September 1963) may not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models that make use of the Dougall-Rohsenow correlation and were approved prior to October 17, 1988, continue to be acceptable until a change is made to, or an error is corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable. For this purpose, a significant reduction in the overall conservatism in the evaluation model would be a reduction in the calculated peak fuel cladding temperature of at least 50 °F from that which would have been calculated on October 17, 1988, due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

39 FR 1001

6. *Pump Modeling.* The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

7. *Core Flow Distribution During Blowdown.* (Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

39 FR 1001

D. POST-BLOWDOWN PHENOMENA; HEAT REMOVAL BY THE ECCS

1. *Single Failure Criterion.* An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

2. *Containment Pressure.* The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

3. *Calculation of Reflood Rate for Pressurized Water Reactors.* The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971; "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7435, January 1970; "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972).

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

4. *Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors.* The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.

5. *Refill and Reflood Heat Transfer for Pressurized Water Reactors.* For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results ("PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied, presently available FLECHT heat transfer correlations ("PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972) are not acceptable. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.

During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer.

6. *Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling.* Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:

a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu/hr-ft²-°F shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu/hr-ft²-°F shall be applied to all fuel rods.

7. *The Boiling Water Reactor Channel Box Under Spray Cooling.* Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7 x 7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu/hr-ft²-°F shall be applied to both sides of the channel box.

c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis (Loss-of-Coolant Ac-

dent and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company Report NEDO-10329, April 1971.

II. REQUIRED DOCUMENTATION

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability—Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this Appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

APPENDIX L—INFORMATION REQUESTED BY THE ATTORNEY GENERAL FOR ANTITRUST REVIEW FACILITY LICENSE APPLICATIONS

Introduction. The information in this appendix is that requested by the Attorney General in connection with his review, pursuant to section 105c of the Atomic Energy Act of 1954, as amended, of certain license applications for nuclear power plants. The applicant shall submit the information as a separate document titled, "Information Requested by the Attorney General for Antitrust Review." Twenty (20) copies shall be submitted prior to any other part of the facility license application as provided in § 50.33a and in accordance with § 2.101 of this chapter.

I. DEFINITIONS

1. "Applicant" means the entity applying for authority to construct or operate subject unit and each corporate parent, subsidiary and affiliate. Where application is made by two or more electric utilities not under common ownership or control, each utility, subject to the applicable exclusions contained in § 50.33a, should set forth separate responses to each item herein.

2. "Subject unit" means the nuclear generating unit or units for which application for construction or operation is being made.

3. "Electric utility" or "system" means any entity owning, controlling or operating facilities for the generation or transmission or distribution of electric power.

4. "Coordination" means any arrangement between two or more systems for generation and transmission planning or operation of two or more interconnected electric utilities not under common ownership or control, including but not limited to arrangements for sharing operating and installed reserves, arrangements for joint or staggered construction of generating facilities, economy energy transactions, capacity transactions based on load diversities, thermal/hydro generation pooling, common maintenance arrangements, and joint use of transmission facilities or wheeling.

5. "Coordinating power and energy" means energy transmitted in accordance with an arrangement for coordination including but not limited to emergency power, economy energy, deficiency power and associated energy, and maintenance power and energy.

6. Except where specifically mentioned otherwise, the term "reserve generating capacity" or "reserves" shall refer to installed reserves in contrast to spinning or operating reserves.

II. REQUIRED INFORMATION

1. State separately for hydroelectric and thermal generating resources applicant's most recent peak load and dependable capacity for the same time period. State applicant's dependable capacity at time of system peak for each of the next 10 years for which information is available. Identify each new unit or resource. For hydroelectric generating capacity, indicate the number of kilowatt hours of use associated with each kilowatt of capacity during the "adverse water year" upon which dependable capacity is based. Indicate average annual kilowatt hour loads per kilowatt, associated with each system peak shown (exclusive of interchange arrangements).

2. State applicant's estimated annual load growth for each of the next 20 years or for the period applicant utilizes in system planning. Indicate growth both in kilowatt requirements and kilowatt hour requirements.

3. State estimated annual load growth in kilowatts and kilowatt hours of companies or pools upon which the economic justification of the subject unit is based for each of the next 20 years or for the period applicant utilized in system planning. Identify each company or pool member.

4. For the year the subject unit would first come on line, state estimated annual load growth in kilowatts and kilowatt hours of any coordinating group or pool of which the applicant is a member (other than the coordinating group or pool referred to in the applicant's response to item 3) which has generating and/or transmission planning functions. Identify each company or pool member whose loads are indicated in the response thereto.

5. State applicant's minimum installed reserve criterion (as a percentage of load)[1] for the period when the subject unit will first come on line. If the applicant shares reserves with other systems, identify the other systems and provide minimum installed reserve criterion (as a percentage of load)[1] by contracting parties or pool for the period when the proposed unit will first come on line.

6. Describe methods used as a basis to establish, or as a guide in establishing the criteria for applicant's and/or applicant's pool's minimum amount of installed reserves (e.g., (a) single largest unit down, (b) probability methods such as loss of load one day in 20 years, loss of capacity once in 5 years, (c) other methods and/or (d) judgment). List contingencies other than risk of forced outage that enter into the determination.

7. Indicate whether applicant's system interconnections are credited explicitly or implicitly in establishing applicant's installed reserves.

8. List rights to receive emergency power and obligations to deliver emergency power, rights or obligations to receive or deliver deficiency power or unit power, or other coordinating arrangements, by reference to applicant's Federal Power Commission (FPC) rate schedules (i.e., ABC Power & Light Co., FPC Rate Schedule No. 15 including supplements 1-5), and also by reference to applicant's state commission filings. Where documents are not on file with the FPC, supply copies, or where not reduced to writing, describe arrangements. Identify for each such arrangement the participating parties other than applicant. Provide one line electrical and geographic diagrams of coordinating groups or power pools (with generation or transmission planning functions) of which applicant's generation and transmission facilities constitute a part.

9. List, and provide the mailing address for, non-affiliated electric utility systems with peak loads smaller than applicant's which serve either at wholesale or at retail adjacent to areas served by the applicant.

Provide a geographic one line diagram of applicant's generating and transmission facilities (including subtransmission) indicating the location of adjacent systems and as to such systems indicate (if available) their load, their annual load growth, their generating capacity, their largest thermal generating unit size, and their minimum reserve criteria.

10. List separately those systems in item 9 which purchase from applicant (a) all bulk power supply and (b) systems which purchase partial bulk power supply requirements. Where information is available to applicant, identify those Item 9 systems purchasing part or all of their bulk power supply requirements from suppliers other than applicant.

11. State as to all power generated and sold by applicant the most recent average cost of bulk power supply experienced by applicant (a) at site of generating facilities, (b) at the delivery points from the primary transmission (backbone) system, (c) at delivery points from the secondary transmission system, and (d) at delivery points from the distribution system, in terms of dollars per kilowatt per year, in mills per kilowatt hour, and in both the kilowatt costs and kilowatt hour costs divided by the kilowatt hours. If wholesale sales are made at varying voltages, indicate average costs at each voltage.

12. State (a) for generating facilities and (b) for transmission sub-divided by voltage classes, the most recent estimated cost of applicant's bulk power supply expansion program of which the subject unit is a part, in terms of dollars per kilowatt per year, in mills per kilowatt hour and in both the kilowatt costs and kilowatt hour costs divided by the kilowatt hours. Also state separately the most recently estimated cost of the subject unit(s).

13. List and describe all requests for, or indications of interest in, interconnection and/or coordination and purchases or sales of coordinating power and energy from adjacent utilities listed in item 9 since 1960 and state applicant's response thereto. List and describe all requests for, or indications of interest in, supply of full or partial requirements of bulk power for the same period and state applicant's response thereto.

[1] Indicate whether loads other than peak loads are considered.

[2] List separately and identify certificates of concurrence.

14. List (a) agreements to which applicant is a party (reproducing relevant paragraphs) and (b) State laws (supply citations only) which restrict or preclude coordination by, with, between, or among any electric utilities or systems identified in applicant's response to items 8 and 9. List (a) agreements to which the applicant is a party (reproducing relevant paragraphs) and (b) State laws (supply citations only) which restrict or preclude substitution of service or establishment of service of full or partial bulk power supply requirements by an electric utility other than applicant to systems identified in items 8 and 9. Where the contract provision appears in contracts or rate schedules on file with a Federal agency, identify each in the same form as in previous responses. Where the contract has not been filed with a Federal agency, a copy should be supplied unless it has been supplied pursuant to another item hereto. Where it is not in writing, it should be described.

15. State, at point of delivery, average future costs of power purchased from applicant to adjacent systems identified in applicant's response to item 9 in terms of dollars/month/kw for capacity, mills/kw for energy and mills/kwh for both power and energy at purchaser's present load factor (a) at present load, (b) at 50 percent increase over present load, (c) at 100 percent increase over present load, and (d) at 200 percent increase over present load. (All costs should be determined under present rate schedules.) Where sales are made under contracts or rate schedules on file with a Federal agency and not included in the response to item 9, identify each in the same form as in previous responses. Where the contract has not been filed with a Federal agency, a copy should be supplied.

16. State whether applicant has prepared, caused to be prepared, or received engineering studies for generation and transmission expansion programs which include loads of each system in item 9.

17. List adjacent systems to which applicant has offered to sponsor or to conduct system surveys in contemplation of an offer by applicant to purchase, merge or consolidate with said adjacent system, subsequent to January 1, 1960.

18. List applicant's offers or proposals to purchase, merge or consolidate with electric utilities, subsequent to January 1, 1960.

19. List all acquisitions of or mergers or consolidations with electric utilities by applicant, subsequent to January 1, 1960, including:

(a) The name and principal place of business of the system prior to the acquisition, merger or consolidation.

(b) The date the acquisition, merger or consolidation was consummated.

(c) Gross annual revenue and most recent peak load, dependable capacity and the largest thermal generating unit of the system, prior to the dates of consummation.

20. State applicant's six (or fewer if there are not six) lowest industrial or large commercial rates for firm electric power supply in terms of cost for power and energy in mills per kilowatt hour (and separately, the demand and energy component(s) and indicate the portion of the charge attributed to bulk power supply. State the rates or rate blocks applicant utilizes for its six (or fewer if there are not six) promotional services such as electric space heating, electric hot water heating, and the like, in terms of mills per kilowatt hour for power and energy and indicate the portion of the rate or rate blocks attributed to bulk power supply.

38 FR 3955

38 FR 30251

APPENDIX M—STANDARDIZATION OF DESIGN; MANUFACTURE OF NUCLEAR POWER REACTORS; CONSTRUCTION AND OPERATION OF NUCLEAR POWER REACTORS MANUFACTURED PURSUANT TO COMMISSION LICENSE

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import, or export any production or utilization facility. The regulations in the part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, and the issuance of an operating license before operation of the facility. The provisions of this part relating to the facility licensing process are, in general, predicated on the assumption that the facility will be assembled and constructed on the site at which it is to be operated. In those circumstances, both facility design and site-related issues can be considered in the initial construction permit stage of the licensing process.

However, under the Atomic Energy Act, a license may be sought and issued authorizing the manufacture of facilities but not their construction and installation at the sites on which the facilities are to be operated. Prior to the "commencement of construction", as defined in § 50.10(c), of a facility (manufactured pursuant to such a Commission license) on the site at which it is to operate—that is preparation of the site and installation of the facility—a construction permit that, among other things, reflects approval of the site on which the facility is to be operated, must be issued by the Commission. This appendix sets out the particular requirements and provisions applicable to such situations where nuclear power reactors to be manufactured pursuant to a Commission license and subsequently installed at the site pursuant to a Commission construction permit, are of the type described in § 50.22. It thus codifies one approach to the standardization of nuclear power reactors.

38 FR 30253

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions in this part applicable to construction permits, including the requirement in § 50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of a public hearing, apply in context, with respect to matters of radiological health and safety, environmental protection, and the common defense and security, to licenses pursuant to this Appendix M to manufacture nuclear power reactors (manufacturing licenses) to be operated at sites not identified in the license application.

43 FR 49303

2. An application for a manufacturing license pursuant to this Appendix M must be submitted, as specified in § 50.4, and meet all the requirements of §§ 50.34(a) (1)–(9) and 50.34a (a) and (b), except that the preliminary safety analysis report shall be designated as a "design report" and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application. The application must also include information pertaining to design features of the proposed reactor(s) that affect plans for coping with emergencies in the operation of the reactor(s).

49 FR 9352

3. An applicant for a manufacturing license pursuant to this Appendix M shall submit with his application an environmental report as required of applicants for construction permits in accordance with Subpart A of Part 51 of this chapter, provided, however, that such report shall be directed at the manufacture of the reactor(s) at the manufacturing site and, in general terms, at the construction and operation of the reactor(s) at an hypothetical site or sites having characteristics that fall within the postulated site parameters. The related draft and final environmental impact statement prepared by the Commission's regulatory staff will be similarly directed.

49 FR 9352

38 FR 30253

4. (a) Sections 50.10 (b) and (c), 50.12(b), 50.23, 50.30(d), 50.34(a)(10), 50.34a(c), 50.35 (a) and (c), 50.40(a), 50.45, 50.55(d), 50.56, and Appendix J do not apply to manufacturing licenses. Appendices E and H apply to manufacturing licenses only to the extent that the requirements of these appendixes involve facility design features.

50 FR 1865

(b) The financial information submitted pursuant to § 50.33(f) and Appendix C shall be directed at a demonstration of the financial qualifications of the applicant for the manufacturing license to carry out the manufacturing activity for which the license is sought.

38 FR 30253

5. The Commission may issue a license to manufacture one or more nuclear power reactors to be operated at sites not identified in the license application if the Commission finds that:

(a) The applicant has described the proposed design of and the site parameters postulated for the reactor(s), including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.

(b) Such further technical or design information as may be required to complete the design report and which can reasonably be left for later consideration, will be supplied in a supplement to the design report.

(c) Safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

(d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved before any of the proposed nuclear power reactor(s) are removed from the manufacturing site and (ii) taking into consideration the site criteria contained in Part 100 of this chapter, the proposed reactor(s) can be constructed and operated at sites having characteristics that fall within the site parameters postulated for the design of the reactor(s) without undue risk to the health and safety of the public.

(e) The applicant is technically and financially qualified to design and manufacture the proposed nuclear power reactor(s).

(f) The issuance of a license to the applicant will not be inimical to the common defense and security or to the health and safety of the public.

49 FR 9352

(g) On the basis of the evaluations and analyses of the environmental effects of the proposed action required by subpart A of Part 51 of this chapter and paragraph 3 of this Appendix, the action called for is the issuance of the license.

Note.—When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the reactor(s), the findings required for the issuance of the license will be appropriately modified to reflect that fact.

6. Each manufacturing license issued pursuant to this appendix will specify the number of nuclear power reactors authorized to be manufactured and the latest date for the completion of the manufacture of all such reactors. Upon good cause shown, the Commission will extend such completion date for a reasonable period of time.

7. The holder of a manufacturing license issued pursuant to this Appendix M shall submit to the Commission the final design of the nuclear power reactor(s) covered by the license as soon as such design has been completed. Such submittal shall be in the form of an application for amendment of the manufacturing license.

8. The prohibition in § 50.10(e) against commencement of construction of a production or utilization facility prior to issuance of a construction permit applies to the transport of a nuclear power reactor(s) manufactured pursuant to a license issued pursuant to this appendix from the manufacturing facility to the site at which the reactor(s) will be installed and operated. In addition, such nuclear power reactor(s) shall not be removed from the manufacturing site until the final design of the reactor(s) has been approved by the Commission in accordance with paragraph 9.

9. An application for a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this Appendix M need not contain such information or analyses as have previously been submitted to the Commission in connection with the application for a manufacturing license, but shall contain, in addition to the information and analyses otherwise required by §§ 50.34(a) and 50.34a, sufficient information to demonstrate that the site on which the reactor(s) is to be operated falls within the postulated site parameters specified in the relevant manufacturing license application.

10. The Commission may issue a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this Appendix M if the Commission (a) finds that the site on which the reactor is to be operated falls within the postulated site parameters specified in the relevant application for a manufacturing license and (b) makes the findings otherwise required by this part. In no event will a construction permit be issued until the relevant manufacturing license has been issued.

[38 FR 30253]

11. An operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this Appendix M may be issued by the Commission pursuant to § 50.57 and Subpart A of Part 51 of this chapter except that the Commission shall find, pursuant to § 50.57(a)(1), that construction of the reactor(s) has been substantially completed in conformity with both the manufacturing license and the construction permit and the applications therefor, as amended, and the provisions of the Act, and the rules and regulations of the Commission. Notwithstanding the other provisions of this paragraph, no application for an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this Appendix M will be docketed until the application for an amendment to the relevant manufacturing license required by paragraph 7 has been docketed.

[49 FR 9352]

12. In making the findings required by this part for the issuance of a construction permit or an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix, or an amendment to such a manufacturing license, construction permit, or operating license, the Commission will treat as resolved those matters which have been resolved at an earlier stage of the licensing process, unless there exists significant new information that substantially affects the conclusion(s) reached at the earlier stage or other good cause.

[38 FR 30253]

APPENDIX N—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OR DUPLICATE DESIGN AT MULTIPLE SITES

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 of this part require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import or export any production or utilization facility. The regulations in this part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, except as provided in § 50.10(e), and the issuance of an operating license before operation of the facility.

The Commission's regulations in Part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings (§ 2.761a, Appendix A, section I(c)), and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§ 2.715a, 2.716).

This Appendix sets out the particular requirements and provisions applicable to situations in which applications are filed by one or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites.[1]

1. Except as otherwise specified in this Appendix or as the context otherwise indicates, the provisions of this part applicable to construction permits and operating licenses, including the requirement in § 50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of public hearings, apply to construction permits and operating licenses subject to this Appendix N.

[40 FR 2975]

2. Applications for construction permits submitted pursuant to this Appendix must include the information required by §§ 50.33, 50.33a, 50.34(a) and 50.34a (e) and (b) and be submitted as specified in § 50.4. The applicant shall also submit the information required by § 51.50 of this chapter. * * *

[51 FR 40303]

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and Subpart D of Part 2 of this chapter.

For the technical information required by §§ 50.34(a) (1) through (5) and (8) and 50.34a (a) and (b), reference may be made to a single preliminary safety analysis of the design[1] which, for the purposes of § 50.34(a)(1) includes one set of site parameters postulated for the design of the reactors, and an analysis and evaluation of the reactors in terms of such postulated site parameters. Such single preliminary safety analysis shall also include information pertaining to design features of the proposed reactors that affect plans for coping with emergencies in the operation of the reactors, and shall describe the quality assurance program with respect to aspects of design, fabrication, procurement and construction that are common to all of the reactors.

3. Applications for operating licenses submitted pursuant to this Appendix N shall include the information required by §§ 50.33, 50.34(b) and (c), and 50.34a(c). The applicant shall also submit the information required by § 51.53 of this chapter. For the technical information required by §§ 50.34(b)(2)–(5) and 50.34a(c), reference may be made to a single final safety analysis of the design.

49 FR 9352

[1] As used in this Appendix, the design of a nuclear power reactor included in a single referenced safety analysis report means the design of those structures, systems and components important to radiological health and safety and the common defense and security.

APPENDIX O—STANDARDIZATION OF DESIGN; STAFF REVIEW OF STANDARD DESIGNS

This appendix sets out procedures for the filing, staff review and referral to the Advisory Committee on Reactor Safeguards of standard designs for a nuclear power reactor of the type described in § 50.22 or major portions thereof.

1. Any person may submit a proposed preliminary or final standard design for a nuclear power reactor of the type described in § 50.22 to the regulatory staff for its review. Such a submittal may consist of either the preliminary or final design for the entire reactor facility or the preliminary or final design of major portions thereof.

2. The submittal for review of the standard design must be made in the same manner and in the same number of copies as provided in §§ 50.4 and 50.30 for license applications.

51 FR 24650

3. The submittal for review of the standard design shall include the information described in § 50.33(a)–(d) and the applicable technical information required by §§ 50.34(a) and (b), as appropriate, and 50.34a (other than that required by §§ 50.34(a)(6) and (10), 50.34(b)(1), (6)(i), (ii), (iv), and (v) and 50.34(b)(7) and (8)). The submittal shall also include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant. With respect to the requirements of §§ 50.34(a)(1), the submittal for review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters. The information submitted pursuant to § 50.34(a)(7) shall be limited to the quality assurance program to be applied to the design, procurement and fabrication fo the structures, systems, and components for which design review has been requested and the information submitted pursuant to § 50.34(a)(9) shall be limited to the qualifications of the person submitting the standard design to design the reactor or major portion thereof. The submittal shall also include information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof.

40 FR 28176

4. Once the regulatory staff has initiated a technical review of a submittal under this Appendix, the submittal will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report.

5. Upon completion of their review of a submittal under this Appendix, the regulatory staff shall publish in the FEDERAL REGISTER a determination as to whether or not the preliminary or final design is acceptable, subject to such conditions as may be appropriate, and make available in the Public Document Room an analysis of the design in the form of a report. An approved design shall be utilized by and relied upon by the regulatory staff and the ACRS in their review of any individual facility license application, which incorporates by reference a design approved in accordance with this paragraph unless there exists significant new information which substantially affects the earlier determination or other good cause.

6. The determination and report by the regulatory staff shall not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Panel, and other presiding officers in any proceeding under Subpart G of Part 2 of this chapter.

40 FR 28176

7. The Commission may, on its own initiative or in response to a petition for rule making, approve the design in a rule making proceeding and in that event, the approved design will be subject to challenge only as provided in § 2.758 of this chapter. An environmental impact statement may be prepared for such a rule making action in accordance with §§ 51.20(b)(13) and 51.85 of this chapter. If an environmental impact statement is prepared, the Commission may require the petitioner for rule making to submit information to the Commission to aid the Commission in the preparation of the environmental impact statement.

8. Information requests to the approval holder regarding an approved design shall be evaluated prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each such evaluation performed by the NRC staff shall be in accordance with 10 CFR 50.54(f) and shall be approved by the Executive Director for Operations or his or her designee prior to issuance of the request.

50 FR 38097

Appendix Q—Pre-Application Early Review of Site Suitability Issues

49 FR 9352

This appendix sets out procedures for the filing, Staff review, and referral to the Advisory Committee on Reactor Safeguards of requests for early review of one or more site suitability issues relating to the construction and operation of certain utilization facilities separately from and prior to the submittal of applications for construction permits for the facilities. The appendix also sets out procedures for the preparation and issuance of Staff Site Reports and for their incorporation by reference in applications for the construction and operation of certain utilization facilities. The utilization facilities are those which are subject to § 51.20(b) of this chapter and are of the type specified in § 50.21(b) (2) or (3) or § 50.22 or are testing facilities. This appendix does not apply to proceedings conducted pursuant to Subpart F of Part 2 of this chapter.

42 FR 22867

1. Any person may submit information regarding one or more site suitability issues to the Commission's Staff for its review separately from and prior to an application for a construction permit for a facility. Such a submittal shall be accompanied by any fee required by Part 170 of this chapter and shall consist of the portion of the information required of applicants for construction permits by §§ 50.33(a)-(c) and (e), and, insofar as it relates to the issue(s) of site suitability for which early review is sought, by §§ 50.34(a)(1) and 50.30(f), except that information with respect to operation of the facility at the projected initial power level need not be supplied.

51 FR 40303

2. The submittal for early review of site suitability issue(s) must be made in the same manner and in the same number of copies as provided in §§ 50.4 and 50.30 for license applications. The submittal must include sufficient information concerning a range of postulated facility design and operation parameters to enable the Staff to perform the requested review of site suitability issues. The submittal must contain suggested conclusions on the issues of site suitability submitted for review and must be accompanied by a statement of the bases or the reasons for those conclusions. The submittal must also list, to the extent possible, any long-range objectives for ultimate development of the site, state whether any site selection process was used in preparing the submittal, describe any site selection process used, and explain what consideration, if any, was given to alternative sites.

42 FR 22882

3. The Staff shall publish a notice of docketing of the submittal in the FEDERAL REGISTER, and shall send a copy of the notice of docketing to the Governor or other appropriate official of the State in which the site is located. This notice shall identify the location of the site, briefly describe the site suitability issue(s) under review, and invite comments from Federal, State, and local agencies and interested persons within 120 days of publication or such other time as may be specified, for consideration by the staff in connection with the initiation or outcome of the review and, if appropriate by the ACRS, in connection with the outcome of their review. The person requesting review shall serve a copy of the submittal on the Governor or other appropriate official of the State in which the site is located, and on the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, on the chief executive of the county. The portion of the submittal containing information required of applicants for construction permits by §§ 50.33(a)-(c) and (e) and 50.34(a)(1) will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report. There will be no referral to the ACRS unless early review of the site safety issues under § 50.34(a)(1) is requested.

53 FR 43419

4. Upon completion of review by the staff and, if appropriate by the ACRS, of a submittal under this appendix, the staff shall prepare a Staff Site Report which shall identify the location of the site, state the site suitability issues reviewed, explain the nature and scope of the review, state the conclusions of the staff regarding the issues reviewed and state the reasons for those conclusions. Upon issuance of a Staff Site Report, the Staff shall publish a notice of the availability of the report in the FEDERAL REGISTER and shall place copies of the report in the Commission's Public Document Room at 2120 L Street, NW, Washington, D.C. 20555, and in a local Public Document Room(s) located near the site identified in the Staff Site Report. The Staff shall also send a copy of the report to the Governor or other appropriate official of the State in which the site is located, and to the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, to the chief executive of the county.

49 FR 9352

5. Any Staff Site Report prepared and issued in accordance with this appendix may be incorporated by reference, as appropriate, in an application for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter and is of the type specified in §§ 50.21(b) (2) or (3) or 50.22 of this chapter or is a testing facility. The conclusions of the Staff Site Report will be reexamined by the staff where five years or more have elapsed between the issuance of the Staff Site Report and its incorporation by reference in a construction permit application.

42 FR 22882

6. Issuance of a Staff Site Report shall not constitute a commitment to issue a permit or license, to permit on-site work under § 50.10(e), or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Board, and other presiding officers in any proceeding under Subpart F and/or G of Part 2 of this chapter.

49 FR 9352

7. The staff will not conduct more than one review of site suitability issues with regard to a particular site prior to the full construction permit review required by Subpart A of Part 51 of this chapter. The staff may decline to prepare and issue a Staff Site Report in response to a submittal under this Appendix where it appears that, (a) in cases where no review of the relative merits of the submitted site and alternative sites under Subpart A of Part 51 of this chapter is requested, there is a reasonable likelihood that further Staff review would identify one or more preferable alternative sites and the Staff review of one or more site suitability issues would lead to an irreversible and irretrievable commitment of resources prior to the submittal of the analysis of alternative sites in the Environmental Report that would prejudice the later review and decision on alternative sites under Subpart F and/or G of Part 2 and Subpart A of Part 51 of this chapter; or (b) in cases where, in the judgment of the Staff, early review of any site suitability issue or issues would not be in the public interest, considering (1) the degree of likelihood that any early findings on those issues would retain their validity in later reviews, (2) the objections, if any, of cognizant state or local government agencies to the conduct of an early review on those issues, and (3) the possible effect on the public interest of having an early, if not necessarily conclusive, resolution of those issues.

APPENDIX R—FIRE PROTECTION PROGRAM FOR NUCLEAR POWER FACILITIES OPERATING PRIOR TO JANUARY 1, 1979

I. INTRODUCTION AND SCOPE

This appendix applies to licensed nuclear power electric generating stations that were operating prior to January 1, 1979, except to the extent set forth in § 50.48(b) of this part. With respect to certain generic issues for such facilities it sets forth fire protection features required to satisfy Criterion 3 of Appendix A to this part.[1]

Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boiloff.

The phrases "important to safety," or "safety-related," will be used throughout this Appendix R as applying to all safety functions. The phrase "safe shutdown" will be used throughout this appendix as applying to both hot and cold shutdown functions.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under postfire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents. Three levels of fire damage limits are established according to the safety functions of the structure, system, or component:

Safety function	Fire damage limits
Hot Shutdown	One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire, including an exposure fire.[2]
Cold Shutdown	Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, including an exposure fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability.
Design Basis Accidents	Both trains of equipment necessary for mitigation of consequences following design basis accidents may be damaged by a single exposure fire.

[2] *Exposure Fire.* An exposure fire is a fire in a given area that involves either in situ or transient combustibles and is external to any structures, systems, or components located in or adjacent to that same area. The effects of such fire (e.g., smoke, heat, or ignition) can adversely affect those structures, systems, or components important to safety. Thus, a fire involving one train of safe shutdown equipment may constitute an exposure fire for the redundant train located in the same area, and a fire involving combustibles other than either redundant train may constitute an exposure fire to both redundant trains located in the same area.

The most stringent fire damage limit shall apply for those systems that fall into more than one category. Redundant systems used to mitigate the consequences of other design basis accidents but not necessary for safe shutdown may be lost to a single exposure fire. However, protection shall be provided so that a fire within only one such system will not damage the redundant system.

II. GENERAL REQUIREMENTS

A. *Fire protection program.* A fire protection program shall be established at each nuclear power plant. The program shall establish the fire protection policy for the protection of structures, systems, and components important to safety at each plant and the procedures, equipment, and personnel required to implement the program at the plant site.

The fire protection program shall be under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position and who has available staff personnel knowledgeable in both fire protection and nuclear safety.

The fire protection program shall extend the concept of defense-in-depth to fire protection in fire areas important to safety, with the following objectives:

- to prevent fires from starting;
- to detect rapidly, control, and extinguish promptly those fires that do occur;
- to provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

B. *Fire hazards analysis.* A fire hazards analysis shall be performed by qualified fire protection and reactor systems engineers to (1) consider potential in situ and transient fire hazards; (2) determine the consequences of fire in any location in the plant on the ability to safely shut down the reactor or on the ability to minimize and control the release of radioactivity to the environment; and (3) specify measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability as required for each fire area containing structures, systems, and components important to safety in accordance with NRC guidelines and regulations.

C. *Fire prevention features.* Fire protection features shall meet the following general requirements for all fire areas that contain or present a fire hazard to structures, systems, or components important to safety.

1. In situ fire hazards shall be identified and suitable protection provided.

2. Transient fire hazards associated with normal operation, maintenance, repair, or modification activities shall be identified and eliminated where possible. Those transient fire hazards that can not be eliminated shall be controlled and suitable protection provided.

3. Fire detection systems, portable extinguishers, and standpipe and hose stations shall be installed.

4. Fire barriers or automatic suppression systems or both shall be installed as necessary to protect redundant systems or components necessary for safe shutdown.

5. A site fire brigade shall be established, trained, and equipped and shall be on site at all times.

6. Fire detection and suppression systems shall be designed, installed, maintained, and tested by personnel properly qualified by experience and training in fire protection systems.

7. Surveillance procedures shall be established to ensure that fire barriers are in place and that fire suppression systems and components are operable.

D. *Alternative or dedicated shutdown capability.* In areas where the fire protection features cannot ensure safe shutdown capability in the event of a fire in that area, alternative or dedicated safe shutdown capability shall be provided.

III. SPECIFIC REQUIREMENTS

A. *Water supplies for fire suppression systems.* Two separate water supplies shall be provided to furnish necessary water volume and pressure to the fire main loop.

Each supply shall consist of a storage tank, pump, piping, and appropriate isolation and control valves. Two separate redundant suctions in one or more intake structures from a large body of water (river, lake, etc.) will satisfy the requirement for two separated water storage tanks. These supplies shall be separated so that a failure of one supply will not result in a failure of the other supply.

Each supply of the fire water distribution system shall be capable of providing for a period of 2 hours the maximum expected water demands as determined by the fire hazards analysis for safety-related areas or other areas that present a fire exposure hazard to safety-related areas.

When storage tanks are used for combined service water/fire water uses the minimum volume for fire uses shall be ensured by means of dedicated tanks or by some physical means such as a vertical standpipe for other water service. Administrative controls, including locks for tank outlet valves, are unacceptable as the only means to ensure minimum water volume.

Other water systems used as one of the two fire water supplies shall be permanently connected to the fire main system and shall be capable of automatic alignment to the fire main system. Pumps, controls, and power supplies in these systems shall satisfy the requirements for the main fire pumps. The use of other water systems for fire protection shall not be incompatible with their functions required for safe plant shutdown. Failure of the other system shall not degrade the fire main system.

B. *Sectional isolation valves.* Sectional isolation valves such as post indicator valves or key operated valves shall be installed in the fire main loop to permit isolation of por-

[1] Clarification and guidance with respect to permissible alternatives to satisfy Appendix A to BTP APCSB 9.5-1 has been provided in four other NRC documents:

"Supplementary Guidance on Information Needed for Fire Protection Evaluation," dated October 21, 1976.

"Sample Technical Specification," dated May 12, 1977.

"Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance," dated June 14, 1977.

"Manpower Requirements for Operating Reactors," dated May 11, 1978.

A Fire Protection Safety Evaluation Report that has been issued for each operating plant states how these guidelines were applied to each facility and identifies open fire protection issues that will be resolved when the facility satisfies the appropriate requirements of Appendix R to Part 50.

46 FR 44734

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tions of the fire main loop for maintenance or repair without interrupting the entire water supply.

C. *Hydrant isolation valves.* Valves shall be installed to permit isolation of outside hydrants from the fire main for maintenance or repair without interrupting the water supply to automatic or manual fire suppression systems in any area containing or presenting a fire hazard to safety-related or safe shutdown equipment.

D. *Manual fire suppression.* Standpipe and hose systems shall be installed so that at least one effective hose stream will be able to reach any location that contains or presents an exposure fire hazard to structures, systems, or components important to safety.

Access to permit effective functioning of the fire brigade shall be provided to all areas that contain or present an exposure fire hazard to structures, systems, or components important to safety.

Standpipe and hose stations shall be inside PWR containments and BWR containments that are not inerted. Standpipe and hose stations inside containment may be connected to a high quality water supply of sufficient quantity and pressure other than the fire main loop if plant-specific features prevent extending the fire main supply inside containment. For BWR drywells, standpipe and hose stations shall be placed outside the dry well with adequate lengths of hose to reach any location inside the dry well with an effective hose stream.

E. *Hydrostatic hose tests.* Fire hose shall be hydrostatically tested at a pressure of 150 psi or 50 psi above maximum fire main operating pressure, whichever is greater. Hose stored in outside hose houses shall be tested annually. Interior standpipe hose shall be tested every three years.

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F. *Automatic fire detection.* Automatic fire detection systems shall be installed in all areas of the plant that contain or present an exposure fire hazard to safe shutdown or safety-related systems or components. These fire detection systems shall be capable of operating with or without offsite power.

G. *Fire protection of safe shutdown capability.* 1. Fire protection features shall be provided for structures, systems, and components important to safe shutdown. These features shall be capable of limiting fire damage so that:

a. One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage; and

b. Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours.

2. Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

c. Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area;

Inside noninerted containments one of the fire protection means specified above or one of the following fire protection means shall be provided:

d. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards;

e. Installation of fire detectors and an automatic fire suppression system in the fire area; or

f. Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield.

3. Alternative or dedicated shutdown capability and its associated circuits,[3] independent of cables, systems or components in the area, room or zone under consideration, shall be provided:

a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or

46 FR 44734

b. Where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems.

In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration.

H. *Fire brigade.* A site fire brigade trained and equipped for fire fighting shall be established to ensure adequate manual fire fighting capability for all areas of the plant containing structures, systems, or components important to safety. The fire brigade shall be at least five members on each shift. The brigade leader and at least two brigade members shall have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The qualification of fire brigade members shall include an annual physical examination to determine their ability to perform strenuous fire fighting activities. The shift supervisor shall not be a member of the fire brigade. The brigade leader shall be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant safety-related systems.

The minimum equipment provided for the brigade shall consist of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus using full-face positive-pressure masks approved by NIOSH (National Institute for Occupational Safety and Health—approval formerly given by the U.S. Bureau of Mines) shall be provided for fire brigade, damage control, and control room personnel. At least 10 masks shall be available for fire brigade personnel. Control room personnel may be furnished breathing air by a manifold system piped from a storage reservoir if practical. Service or rated operating life shall be a minimum of one-half hour for the self-contained units.

At least a 1-hour supply of breathing air in extra bottles shall be located on the plant site for each unit of self-contained breathing appratus. In addition, an onsite 6-hour supply of reserve air shall be provided and arranged to permit quick and complete replenishment of exhausted air supply bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used and the compressors shall be operable assuming a loss of offsite power. Special care must be taken to locate the compressor in areas free of dust and contaminants.

1. *Fire brigade training.* The fire brigade training program shall ensure that the capability to fight potential fires is established and maintained. The program shall consist of an initial classroom instruction program followed by periodic classroom instruction, fire fighting practice, and fire drills:

i. *Instruction*

a. The initial classroom instruction shall include:

(1) Indoctrination of the plant fire fighting plan with specific identification of each individual's responsibilities.

(2) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant.

(3) The toxic and corrosive characteristics of expected products of combustion.

(4) Identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to each area.

(5) The proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), and record file fires.

(6) The proper use of communication, lighting, ventilation, and emergency breathing equipment.

(7) The proper method for fighting fires inside buildings and confined spaces.

(8) The direction and coordination of the fire fighting activities (fire brigade leaders only).

(9) Detailed review of fire fighting strategies and procedures.

(10) Review of the latest plant modifications and corresponding changes in fire fighting plans.

NOTE: Items (9) and (10) may be deleted from the training of no more than two of the non-operations personnel who may be assigned to the fire brigade.

b. The instruction shall be provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant.

c. Instruction shall be provided to all fire brigade members and fire brigade leaders.

[3] Alternative shutdown capability is provided by rerouting, relocating or modification of existing systems; dedicated shutdown capability is provided by installing new structures and systems for the function of post-fire shutdown.

d. Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

e. Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a two-year period. These sessions may be concurrent with the regular planned meetings.

2. *Practice*

Practice sessions shall be held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions shall provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting. These practice sessions shall be provided at least once per year for each fire brigade member.

3. *Drills*

a. Fire brigade drills shall be performed in the plant so that the fire brigade can practice as a team.

b. Drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year.

A sufficient number of these drills, but not less than one for each shift fire brigade per year, shall be unannounced to determine the fire fighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Persons planning and authorizing an unannounced drill shall ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills shall not be scheduled closer than four weeks.

At least one drill per year shall be performed on a "dark shift" for each shift fire brigade.

c. The drills shall be preplanned to establish the training objectives of the drill and shall be critiqued to determine how well the training objectives have been met. Unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance shall be followed by a repeat drill within 30 days.

d. At 3-year intervals, a randomly selected unannounced drill must be critiqued by qualified individuals independent of the licensee's staff. A copy of the written report from these individuals must be available for NRC review and shall be retained as a record as specified in § III.L.4 of this appendix.

e. Drills shall as a minimum include the following:

(1) Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, and selection, placement and use of equipment, and fire fighting strategies.

(2) Assessment of each brigade member's knowledge of his or her role in the fire fighting strategy for the area assumed to contain the fire. Assessment of the brigade member's conformance with established plant fire fighting procedures and use of fire fighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable.

(3) The simulated use of fire fighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming loss of automatic suppression capability.

(4) Assessment of brigade leader's direction of the fire fighting effort as to thoroughness, accuracy, and effectiveness.

4. *Records*

Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadened training for fire fighting within buildings shall be scheduled for all those brigade members whose performance records show deficiencies.

J. Emergency lighting. Emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto.

K. Administrative controls. Administrative controls shall be established to minimize fire hazards in areas containing structures, systems, and components important to safety. These controls shall establish procedures to:

1. Govern the handling and limitation of the use of ordinary combustible materials, combustible and flammable gases and liquids, high efficiency particulate air and charcoal filters, dry ion exchange resins, or other combustible supplies in safety-related areas.

2. Prohibit the storage of combustibles in safety-related areas or establish designated storage areas with appropriate fire protection.

3. Govern the handling of and limit transient fire loads such as combustible and flammable liquids, wood and plastic products, or other combustible materials in buildings containing safety-related systems or equipment during all phases of operating, and especially during maintenance, modification, or refueling operations.

4. Designate the onsite staff member responsible for the inplant fire protection review of proposed work activities to identify potential transient fire hazards and specify required additional fire protection in the work activity procedure.

5. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, brazing, or soldering operations. A separate permit shall be issued for each area where work is to be done. If work continues over more than one shift, the permit shall be valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown.

6. Control the removal from the area of all waste, debris, scrap, oil spills, or other combustibles resulting from the work activity immediately following completion of the activity, or at the end of each work shift, whichever comes first.

7. Maintain the periodic housekeeping inspections to ensure continued compliance with these administrative controls.

8. Control the use of specific combustibles in safety-related areas. All wood used in safety-related areas during maintenance, modification, or refueling operations (such as lay-down blocks or scaffolding) shall be treated with a flame retardant. Equipment of supplies (such as new fuel) shipped in untreated combustible packing containers may be unpacked in safety-related areas if required for valid operating reasons. However, all combustible materials shall be removed from the area immediately following the unpacking. Such transient combustible material, unless stored in approved containers, shall not be left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material such as wood or paper excelsior, or polyethylene sheeting shall be placed in metal containers with tight-fitting self-closing metal covers.

9. Control actions to be taken by an individual discovering a fire, for example, notification of control room, attempt to extinguish fire, and actuation of local fire suppression systems.

10. Control actions to be taken by the control room operator to determine the need for brigade assistance upon report of a fire or receipt of alarm on control room annunciator panel, for example, announcing location of fire over PA system, sounding fire alarms, and notifying the shift supervisor and the fire brigade leader of the type, size, and location of the fire.

11. Control actions to be taken by the fire brigade after notification by the control room operator of a fire, for example, assembling in a designated location, receiving directions from the fire brigade leader, and discharging specific fire fighting responsibilities including selection and transportation of fire fighting equipment to fire location, selection of protective equipment, operating instructions for use of fire suppression systems, and use of preplanned strategies for fighting fires in specific areas.

12. Define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. These strategies shall designate:

a. Fire hazards in each area covered by the specific prefire plans.

b. Fire extinguishants best suited for controlling the fires associated with the fire hazards in that area and the nearest location of these extinguishants.

c. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. All access and egress routes that involve locked doors should be specifically identified in the procedure with the appropriate precautions and methods for access specified.

d. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical systems in the zone covered by the specific fire fighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

e. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling should be designated.

f. Organization of fire fighting brigades and the assignment of special duties according to job title so that all fire fighting functions are covered by any complete shift personnel complement. These duties include command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishant to the fire, communication with the control room, and coordination with outside fire departments.

g. Potential radiological and toxic hazards in fire zones.

h. Ventilation system operation that ensures desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

i. Operations requiring control room and shift engineer coordination or authorization.

j. Instructions for plant operators and general plant personnel during fire.

L. *Alternative and dedicated shutdown capability.* 1. Alternative or dedicated shutdown capability provided for a specific fire area shall be able to: (a) achieve and maintain subcritical reactivity conditions in the reactor; (b) maintain reactor coolant inventory; (c) achieve and maintain hot standby conditions for a PWR (hot shutdown[1] for a BWR); (d) achieve cold shutdown conditions within 72 hours; and (e) maintain cold shutdown conditions thereafter. During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, or rupture of the containment boundary.

2. The performance goals for the shutdown functions shall be:

a. The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

b. The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs and be within the level indication in the pressurizer for PWRs.

c. The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

d. The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

e. The supporting functions shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions.

3. The shutdown capability for specific fire areas may be unique for each such area, or it may be one unique combination of systems for all such areas. In either case, the alternative shutdown capability shall be independent of the specific fire area(s) and shall accommodate postfire conditions where offsite power is available and where offsite power is not available for 72 hours. Procedures shall be in effect to implement this capability.

4. If the capability to achieve and maintain cold shutdown will not be available because of fire damage, the equipment and systems comprising the means to achieve and maintain the hot standby or hot shutdown condition shall be capable of maintaining such conditions until cold shutdown can be achieved. If such equipment and systems will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. The number of operating shift personnel, exclusive of fire brigade members, required to operate such equipment and systems shall be on site at all times.

5. Equipment and systems comprising the means to achieve and maintain cold shutdown conditions shall not be damaged by fire; or the fire damage to such equipment and systems shall be limited so that the systems can be made operable and cold shutdown can be achieved within 72 hours. Materials for such repairs shall be readily available on site and procedures shall be in effect to implement such repairs. If such equipment and systems used prior to 72 hours after the fire will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. Equipment and systems used after 72 hours may be powered by offsite power only.

6. Shutdown systems installed to ensure postfire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage.

7. The safe shutdown equipment and systems for each fire area shall be known to be isolated from associated non-safety circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment. The separation and barriers between trays and conduits containing associated circuits of one safe shutdown division and trays and conduits containing associated circuits or safe shutdown cables from the redundant division, or the isolation of these associated circuits from the safe shutdown equipment, shall be such that a postulated fire involving associated circuits will not prevent safe shutdown.[2]

M. *Fire barrier cable penetration seal qualification.* Penetration seal designs shall utilize only noncombustible materials and shall be qualified by tests that are comparable to tests used to rate fire barriers. The acceptance criteria for the test shall include:

1. The cable fire barrier penetration seal has withstood the fire endurance test without passage of flame or ignition of cables on the unexposed side for a period of time equivalent to the fire resistance rating required of the barrier;

2. The temperature levels recorded for the unexposed side are analyzed and demonstrate that the maximum temperature is sufficiently below the cable insulation ignition temperature; and

3. The fire barrier penetration seal remains intact and does not allow projection of water beyond the unexposed surface during the hose stream test.

N. *Fire doors.* Fire doors shall be self-closing or provided with closing mechanisms and shall be inspected semiannually to verify that automatic hold-open, release and closing mechanisms and latches are operable.

One of the following measures shall be provided to ensure they will protect the opening as required in case of fire:

1. Fire doors shall be kept closed and electrically supervised at a continuously manned location;

2. Fire doors shall be locked closed and inspected weekly to verify that the doors are in the closed position;

3. Fire doors shall be provided with automatic hold-open and release mechanisms and inspected daily to verify that doorways are free of obstructions; or

4. Fire doors shall be kept closed and inspected daily to verify that they are in the closed position.

The fire brigade leader shall have ready access to keys for any locked fire doors.

Areas protected by automatic total flooding gas suppression systems shall have electrically supervised self-closing fire doors or shall satisfy option 1 above.

O. *Oil collection system for reactor coolant pump.* The reactor coolant pump shall be equipped with an oil collection system if the containment is not inerted during normal operation. The oil collection system shall be so designed, engineered, and installed that failure will not lead to fire during normal, design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake.[3]

Such collection systems shall be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump lube oil systems. Leakage shall be collected and drained to a vented closed container that can hold the entire lube oil system inventory. A flame arrester is required in the vent if the flash point characteristics of the oil present the hazard of fire flashback. Leakage points to be protected shall include lift pump and piping, overflow lines, lube oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and lube oil reservoirs where such features exist on the reactor coolant pumps. The drain line shall be large enough to accommodate the largest potential oil leak.

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[1] As defined in the Standard Technical Specifications.

[2] An acceptable method of complying with this alternative would be to meet Regulatory Guide 1.75 position 4 related to associated circuits and IEEE Std 384-1974 (Section 4.5) where trays from redundant safety divisions are so protected that postulated fires affect trays from only one safety division.

[3] See Regulatory Guide 1.29—Seismic Design Classification, Paragraph C.2.

REFERENCE PUBLICATIONS

Document Number	Document Title
Un-numbered Lists	Regulatory Guides
NUREG-1642 Revision 1	A Review of NRC Regulatory Processes and Functions
Regulatory Guide 1.70 Rev. 3	Standard Format and Safety Analysis Report for Nuclear Power Plants (LWR Edition)
Regulatory Guide 1.28 Revision 2, and Revision 3	Quality Assurance Program Requirements (Design and Construction), and ASME Codes
Regulatory Guide 1.88 Revision 2	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, and ASME Codes
NUREG-0660, Volumes 1 and 2, Revision 1	NRC Action Plan Developed as a Result of the TMI 2 Accident
NUREG-0737, and Supplement 1	Clarification of TMI Action Plan Requirements
NUREG-0546	Technical Specifications
Regulatory Guide 1.16 Revision 4	Reporting of Operating Information Appendix A, Technical Specifications
Regulatory Guide 1.21 Revision 1	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants
Regulatory Guide 4.1 Revision 1	Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants
NUREG-0472 Revision 3	Draft Radiological Effluent Technical Specifications for PWR's
NUREG-0473 Revision 2	Draft Radiological Effluent Technical Specifications for BWR's
NUREG-0906	Guidance for Implementation of 10 CFR 50.34(g)

REFERENCE PUBLICATIONS (Continued)

Document Number	Document Title
Regulatory Guide 1.9	Backfit Analysis
Issued by letter Dated November 27, 1979 from W. Gammill, NRC, to All Power Reactor Licensees	Branch Technical Position, Revision 1, dated November 1979 (Radiological Assessment)
NUREG-0161	Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File
Regulatory Guide 4.8	Environmental Technical Specifications for Nuclear Power Plants
NUREG-0713 Volume 1	Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1979
NUREG-0654 Revision 1	Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants
NUREG-0452 Revision 4	Standard Technical Specifications for Westinghouse Pressurized Water Reactors
NUREG-0212 Revision 2	Standard Technical Specification (STS) for Combustion Engineering Pressurized Water Reactors (PWR)
NUREG-0103 Revision 4	STS for Babcock and Wilcox PWR
NUREG-0123 Revision 3	STS for Boiling Water Reactors (BWR/5)
NUREG-0799 For Comment	Draft Criteria for Preparation of Emergency Operating Procedures
NUREG-0800	Standard Review Plan
NUREG-0906	Guidance for Implementation of Standard Review Plan Rule 50.34(g)
Regulatory Guide 1.99 Revision 2	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
	CRGR Charter

REFERENCE PUBLICATIONS (Continued)

Document Number	Document Title
NUREG-1070	Severe Accident Policy
NUREG-1109	Regulatory Analysis for the Resolution of USI A-44, Station Blackout (for comment)
NUREG-0588 Revision 1	Interim Staff Position on Environmental Qualification (EQ) of Safety-Related Electrical Equipment.
Regulatory Guide 1.89 Revision 1	EQ of Safety-Related Electrical Equipment
Regulatory Guide	Guidance and Acceptance Criteria regarding the Pressurized Thermal Shock Rule, 10 CFR 50.61
NUREG-1055	Improving Quality and the Assurance of Quality in the design and Construction of Nuclear Power Plants
NUREG-0844	NRC Integrated Program for the Resolution of Unresolved Safety issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity
NUREG-1061 Volumes 1-5	Report of the U.S. Nuclear Regulatory Commission Piping Review Committee
NUREG/CR-3226, 2989, 3992	Technical Reports by NRC consultants at Sandia National Lab and Oak Ridge National Laboratory with respect to the proposed Station Blackout rule.
NUREG-1032	Evaluation of Station Blackout Accidents at Nuclear Power Plants
NUREG-1109	Regulatory Analysis for the Resolution of USI A-44, Station Blackout
NUREG-0459	Generic Adversary Characteristics Summary Report, U.S. NRC March 1979

REFERENCE PUBLICATIONS (Continued)

Document Number	Document Title
Regulatory Guide, DG-1005	Standard Format and Content for Decommissioning Plans for Nuclear Reactors
Regulatory Guide, DG-1006	Records Important for Decommissioning of Nuclear Reactors
NUREG-0654	FEMA-REP-1
NUREG-0844	Steam Generator Tube Integrity, USI A-3, A-5
Reg Guide 2.6 and NUREG-0849	Emergency Planning, Universities/Colleges
Reg. Guides 1.58, 1.64, 1.74, 1.123 1.144, 1.46	10CFR 50.55a, ASME Codes
NUREG-1275, Vol 3.	Operating Experience Feedback Report-Service Water System Failures and Degradation
NUREG-1221	Decommissioning Rule
NUREG-0586	Generic Environmental Statement on Decommissioning
NUREG/CR-0672	PNL Generic Evaluation on Decommissioning, Addenda to NUREG/CR 0130
NUREG-0406	Alternate No. 3, Assessment of BWR Mitigation of ATWS
NUREG-1174	Evaluation of Systems Interactions in Nuclear Power Plants
NUREG-1217	Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants, USI A-47
NUREG-1218	Regulatory Analysis (for USI A-47)
NUREG-1229	Regulatory Analysis for Resolution of USI A-17, Systems Interactions
NUREG-1335	Individual Plant Examination: Submittal Guidance (Policy Statement on Severe Accidents)

REFERENCE PUBLICATIONS (Continued)

Document Number	Document Title
NUREG/CR-5132	Severe Accident Insights Report
NUREG/CR-4920, Volumes 1-5	Assessment of Severe Accident Prevention and Mitigation Features
NUREG/CR-2300	PRA Procedures Guide
NUREG-1175	NRC Safety Research in Support of Regulation
NUREG-1266	NRC Safety Research in Support of Regulation
NUREG-1317	Public Comments on License Renewal (Plant Life Extension)
NUREG-1256	Incentive Regulation of Nuclear Power Plants by State Public Utility Commissions
NUREG-1144	Nuclear Plant Aging Research
NUREG-1147	Seismic Safety Research Program Plan
NUREG-1210	Pilot Program: NRC Severe Reactor Accident Incident Response Training Manual
NUREG-1211	Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualifications of Equipment in Operating Plants
NUREG-1230 (Draft for Comment)	Compendium of ECCS Research for Realistic LOCA Analysis
NUREG-1275	Operating Experience Feedback
NUREG-1281	Evaluation of the Qualification of Spent Fuel for use in Non-Power Reactors
NUREG-1285	"Reed Report" - NRC Staff Evaluation of the GE Company Nuclear Reactor Study
NUREG-1291	BWR and PWR Off-Normal Event Descriptions

REFERENCE PUBLICATIONS (Continued)

Document Number	Document Title
NUREG-1296	Thermal Overload Protection for Electric Motors on Safety Related Motor Operated Valves
NUREG-1289	Shutdown Decay Heat Removal Requirements
NUREG-1273	Containment Integrity Check
NUREG-1266	NRC Safety Research in Support of Regulation
NUREG-1228	Source Term Estimation During Incident Response
NUREG-1233 (Draft for comment)	Seismic Design Criteria
NUREG-1317	Public Comments on License Renewal (Plant Life Extension)
Regulatory Guide 8.12, Revision 2	Criticality Accident Alarm Systems
NUREG/CR-3226, -2998, -3992	NRC Contractors, (Sandia and Oak Ridge National Laboratory) Reports on Station Blackout
NUREG-1409	Backfitting
NUREG-1417	Mark III Containment
Regulatory Guide 1.154	"Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors"
Regulatory Guide 3.62	Onsite Storage of Spent Fuel in Casks at a Nuclear Reactor Site
Regulatory Guide 1.158	Qualification of Safety-Related Lead Storage Batteries, IEEE Standard 535-1986
NUREG-1233	USI A-40, Seismic Design Criteria
NUREG-1316	Technical Findings and Regulatory Analysis Related to GI-70, "Evaluation of Power-Operated Relief Valve and Block Valve

Reliability in PWR Nuclear Power Plants"

NUREG-1326

Regulatory Analysis for the Resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors"