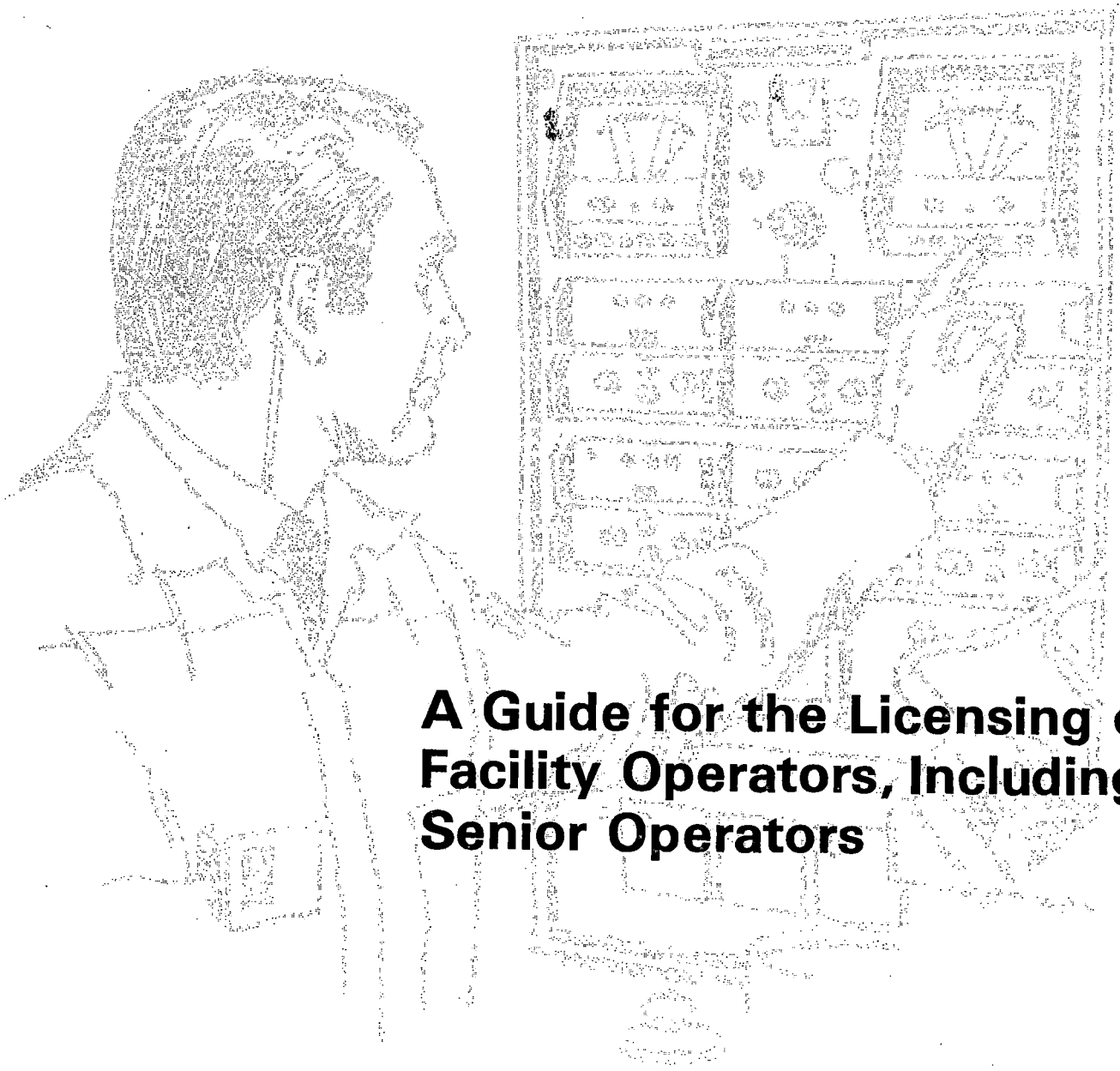
NUREG-0094
Rev. 1 of WASH 1094

NRC Operator Licensing Guide



A Guide for the Licensing of Facility Operators, Including Senior Operators

Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission,
Washington, D.C. 20555

Available from
National Technical Information Service
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Price: Printed Copy $5.50; Microfiche $2.25

A GUIDE FOR THE LICENSING OF FACILITY OPERATORS, INCLUDING SENIOR OPERATORS

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Manuscript Completed: February 1976
Date Published: July 1976

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Washington, D.C. 20555

PREFACE

This guide describes the procedures and criteria for the issuance of operator and senior operator licenses and is intended to assist applicants and facility licensees to better understand the pertinent provisions of the Commission's regulations in this regard. This guide is not a substitute for the regulations and compliance with the guide is not required.

This guide will be revised periodically, as appropriate, to accommodate comments and reflect new information or experience.

Applicable Regulations

For convenience, the following regulations are included as Appendices A, B, & C to this guide.

A - 10 CFR Part 55 - "Operators' Licenses"
B - 10 CFR Part 50 - "Licensing of Production and Utilization Facilities"
C - 10 CFR Part 20 - "Standards for Protection Against Radiation"

Amendments to these regulations may be published from time to time in the Federal Register. It is not intended that the appendices of this guide will be revised each time the regulations are amended. Users of this guide should determine the current status of these regulations.

Copies of Commission regulations may be obtained by addressing a request to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or to any of the U.S. Nuclear Regulatory Commission Regional Offices.

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APPENDICES

I. INTRODUCTION

The Atomic Energy Act of 1954, as amended and incorporated in the Energy Reorganization Act of 1974, requires the Nuclear Regulatory Commission to:

1. Prescribe uniform conditions for licensing individuals as operators of licensed production and utilization facilities;

2. Determine the qualifications of such individuals; and

3. Issue licenses to such individuals.

The basic Commission regulation which implements these requirements of the Act is 10 CFR Part 55. The primary purpose of this guide is to assist the reader in understanding the content and the administration of that regulation.

The regulation concerning licensing of production and utilization facilities, 10 CFR Part 50, contains sections regarding the presence and availability of licensed operators and senior operators at such facilities.

At this time, most existing production and utilization facilities, which are licensed for operation under the regulations, are reactor facilities and, therefore, this guide is principally addressed to this type of facility. It should be noted, however, that the definitions of production and utilization facilities (Section 50.2 of Part 50) include facilities other than reactors.

II. CONTENTS OF APPLICATIONS

An application for a license should contain all of the information set forth in Section 55.10(a) of Part 55. All items of the application, including transmittal letters and supplementary certifications, should be submitted in triplicate EXCEPT the Certificate of Medical Examination, of which only the original is required. The application should contain the following information:

1. The full name, citizenship, age, address and present employment. The address should be the applicant's home mailing address. The present employment should include the applicant's position title and the name and address of the organization by which he is employed.

2. The education and pertinent experience of the applicant. The educational data should be a brief listing of all formal education, including the name of the institution, dates of

attendance, subject or major field of study and type of certificate or diploma, if one was awarded. Technical institute participation should also be listed, as well as military schools in subjects which are related to reactor operation (e.g., electronics, physics, radiological safety, etc.). The experience record should include position titles, employers, locations, dates of employment, and a resume concerning the applicant's duties and responsibilities, with particular emphasis placed upon any prior experience in reactor operations. The amount of detail required will vary in accordance with the complexity of the duties performed and the descriptiveness of the position title. For an individual making an application for a license at a nuclear power plant, the education and experience should clearly indicate that he meets the recommendations of Regulatory Guide 1.8 for the type of license for which he is making application. In general, much greater detail should be provided if consideration for partial or complete waiver of examinations (reference Section 55.24 of 10 CFR Part 55) is being requested.

3. Serial numbers of any operator or senior operator licenses issued by the Commission to the applicant and the expiration date of each.

4. The specific facility (or facilities) for which the applicant seeks an operator or senior operator license. The proper facility name and facility license number should be indicated for each facility.

5. The written request by an authorized representative of the facility licensee that the operating test be administered to the applicant. Even though a request for waiver of examination may be made by an applicant who is experienced in a comparable facility, or by a senior operator license applicant who holds an operator license at the same facility, the request for an operating test should be included so that the examination may be expeditiously scheduled in the event that the request for waiver is denied.

6. Evidence that the applicant has learned to operate the controls in a competent and safe manner and has a need for an operator or a senior operator license in the performance of his duties. The evidence concerning operation of the controls encompasses more than just the aspect of physical manipulation. It should include a certification that the applicant has completed the training required by the facility licensee and has demonstrated, to the satisfaction of the facility licensee, his ability to operate the controls in a competent and safe manner commensurate with the class of license for which he is applying. In addition, the certification should include details on the courses of instruction administered by the facility licensee, number of course hours, number of hours of training and nature of training received at the facility and, for reactors, the startup and shutdown experience received. Applicants must have manipulated the controls of the reactor through at least two reactor startups and have participated as a member in the control room in several other plant transients to be eligible for examination, or have successfully completed an approved training program using a simulator to meet the manipulation requirements. This training breakdown, although generally applicable, may be modified with respect to facilities where the nature of the training program is such that some other form of quantitative description is more illustrative. For a large

group of applicants who have participated in essentially the same training program, the program may be submitted separately in triplicate. Individual applications then need only reference the applicable portions of the training program.

The need for the class of license applied for may be established by a certification of a representative of the facility licensee. In the case of an operator license, it is sufficient that the applicant, after licensing, is expected to be at the facility for a reasonable period of time and that his licensed services will be used, at least part time. In the case of a senior operator license, it is sufficient that the facility licensee desires that the applicant possess such a license, that he is expected to be at the facility for a reasonable period of time, and that his services either to operate the controls or to direct the licensed activities of licensed operators will be used: it is not necessary that he presently be in a position to direct other operators.

7. The need for a license, in order to be eligible for precritical examination at another planned, comparable facility, also constitutes sufficient justification. Refer to item XI below.

8. A report of a current medical examination by a licensed medical practitioner, in the form prescribed in Section 55.60 of Part 55. This form, NRC-396, is attached as Appendix D to this guide. Normally, a medical examination completed within 6 months of the date of the application is considered current.

III. ESTABLISHMENT OF AN EXAMINATION SCHEDULE

A. EXAMINATIONS AT OPERATING REACTORS

Facilities are encouraged to initiate correspondence approximately 45 days in advance of applications, in order to establish tentative examination dates. Otherwise, the Commission will proceed to schedule examinations following receipt and review of applications.

Upon receipt of the applications from personnel of a facility, the Certificates of medical examination will be reviewed by the Commission's medical staff and the medical eligibility of each applicant will be determined (also see Section 55.41 of 10 CFR Part 55).

The Commission will contact the facility management to schedule examinations (refer to Appendix F). In general, when there is a large group of applicants from a facility, it is preferable to schedule the written examinations a few weeks in advance of the operating tests. For small groups, the written and operating tests are usually given consecutively. For examinations involving large groups of applicants, scheduling of time for the operating tests will depend mainly upon the type and physical arrangement of the facility and upon the number of examiners participating.

The most efficient manner of administering examinations to a group of applicants at a facility is to examine them all at one time rather than to examine smaller groups at separate times over a period of months.

B. EXAMINATIONS PRIOR TO INITIAL CRITICALITY

Normally, the Commission schedules the written examination six weeks prior to the projected fuel loading date and the operating test two weeks prior to the projected fuel loading date.

Applications and medicals should be received by the Commission two months prior to the proposed date for the written examinations. At the same time, all normal and off-normal operating procedures, emergency procedures and applicable administrative procedures should be submitted. Determination of the examination dates is dependent, to a large degree, on the timely receipt of the procedures. Facilities should initiate correspondence well in advance of the applications in order to establish tentative examination dates. Refer to item XI below.

IV. CONTENT OF OPERATOR WRITTEN EXAMINATION

The content of the operator written examination is set forth in Section 55.21 of Part 55. That section lists 12 topics which will be included in the examination to the extent applicable at the facility under consideration.

Although it is desirable to separate the written examination into categories in order to determine relative strengths and weaknesses of the applicant, some of the 12 topics listed in the regulation are so closely interrelated that division into 12 separate examination categories is impractical. Therefore, the topics listed in Section 55.21 of Part 55 have been regrouped, for examination purposes, into seven categories which are listed and described below. Sample questions pertaining to each of these categories appear in Appendix E.

A. PRINCIPLES OF REACTOR OPERATION

This category contains questions relating to basic nuclear reactor behavior, elementary nuclear reactor theory, technical terminology and an appreciation of processes taking place in a reactor. Answering these questions requires neither mathematical ability in excess of ordinary algebra nor detailed and advanced knowledge in reactor physics. Questions in this category are about reactors in general or reactors of the appropriate class.

B. FEATURES OF FACILITY DESIGN

This category contains questions about the design features of the particular facility, with emphasis on the reactor and auxiliary systems. The applicant should be able to reproduce, from memory, fairly detailed sketches or descriptions of various hydraulic, pneumatic or electrical distribution systems or of the reactor vessel and core components. Questions are asked about design intent and the more important design parameters. Generally, parameters expressed as limits (e.g., maximum flow, minimum vacuum, maximum pressure, allowable load) or fixed numerical values for fabrication (e.g., enrichment, dimensions) are subjects for questions in this category.

C. GENERAL OPERATING CHARACTERISTICS

This category contains questions on controlled and variable parameters of the reactor and auxiliary systems. Values which are expressed as normal or operating parameters (e.g., purification

flow rate, pressurizer temperature, storage tank level) or values which are measured as resultant characteristics (e.g., temperature coefficient, reactivity worth, pressure drop) are asked for in this category. Questions about the way in which power, reactivity, rod worths, or other parameters of this facility vary in response to rod manipulations, heatup, core burn up, experiment insertion or other stimuli are also in this category. Further included are questions relating to the traces that one would see on recorders during normal and abnormal transients, with the emphasis on facility behavior rather than instrument characteristics. Secondary system transients that induce reactor transients are also subject for questions in this category.

D. INSTRUMENTS AND CONTROLS

This category contains questions on the characteristics and interrelationships of the nuclear and process instrumentation and control systems. These questions will inquire into the principles of operation of detectors, location and setpoints of instruments, diagrammatic representation of instrument and control systems and details of control rod drives design and operation. An applicant is not expected to have the knowledge of an instrument technician but his answers should indicate the ability to recognize the indications and consequences of improper instrument performance (e.g., over-compensation, power failure, air supply failure, signal failure) including the traces that recorders would show. He should also be able to make use of all available instrumentation to provide checks or verification of observed readings.

E. SAFETY AND EMERGENCY SYSTEMS

This category contains questions on the design, construction, operation and interrelationships of the systems most directly associated with reactor safety, such as reactor trip and other power reduction systems, pressure relief, suppression and containment, poison systems, spray systems, emergency power systems, fire protection systems and systems that are intended to mitigate the consequences of the accidental release of radioactive materials. Annunciated malfunctions are also covered in this category. The applicant should demonstrate thorough knowledge of detailed design, characteristics, and operating methods for these systems. He should also be familiar with the conditions which require the use of such systems and the reasons why such protection is required.

F. STANDARD AND EMERGENCY OPERATING PROCEDURES

This category contains questions on the procedures for the operation of the reactor and auxiliary systems, including administrative controls and technical specifications. In general, an applicant must demonstrate complete understanding of the symptoms, automatic actions and immediate action steps specified by off-normal or emergency operating procedures. The applicant should be able to describe generally the objectives and methods used in the normal, off-normal and emergency operating procedures including how to perform the manipulations or verifications. Operating restrictions and limitations in the facility license, including technical specifications, may be included, to the extent they are directly applicable to an operator.

G. RADIATION CONTROL AND SAFETY

This category contains questions on terminology, radiation hazards, radiological safety practices, fixed and portable radiation monitoring equipment and applicable facility procedures regarding discharging and monitoring radioactive releases. The applicant should demonstrate knowledge of the type and magnitude of radiation hazards which might be expected to be present and of measures to cope with them. He should know facility regulations and the general provisions and precautionary procedures of 10 CFR Part 20. He should be able to understand and use portable equipment and describe type, location, approximate range and alarms associated with fixed equipment. He should know the limitations as well as the applications of this equipment.

V. CONTENT OF SENIOR OPERATOR WRITTEN EXAMINATION

As set forth in Section 55.22 of Part 55, the senior operator written examination will contain, to the extent applicable at the facility, nine topics in addition to those specified for an operator. In practice, these topics are grouped into five categories on the examination, which are listed and described below. Sample questions pertaining to each of these categories appear in Appendix E.

H. REACTOR THEORY

This category contains questions on principles of reactor theory including details of the fission process, neutron multiplication, source and control rod effects and criticality indications. It has more advanced content than the operator category A, but is not advanced to the level of a nuclear physicist or engineer. The applicant should be able to demonstrate quantitative as well as qualitative knowledge of reactor behavior. He should be able to understand and use mathematical expressions regarding reactor behavior; however, these expressions (or formulae) and nuclear constants (fission factors, half lives, etc.) usually need not be committed to memory and will be supplied in the examination when questions requiring them are included. Further, this category may contain questions, as applicable to the facility, concerning some aspects of basic reactor engineering (e.g., heat transfer and fluid flow) which affect the safety of the reactor core and vessel.

The primary emphasis throughout will be on understanding and practical application of the theory rather than mere memorization of technical facts.

I. RADIOACTIVE MATERIAL HANDLING, DISPOSAL AND HAZARDS

This category contains questions on radiation hazards which may arise during operation or the performance of experiments, shielding alterations or maintenance activities. Close familiarity with the provisions of 10 CFR Part 20 and supplementary facility regulations is required as well as a good common sense approach to radiological safety situations. Questions may include calculations involving inverse square law, decay rates, half-value thicknesses and conversions of measured radiation intensities to rem, as well as other calculations of a similar nature. Here, operational "Rules of Thumb" methods of calculations are acceptable wherever applicable.

Also included are questions relating to procedures and equipment (processing and monitoring) available for handling and disposal of radioactive materials and effluents. The senior operator should have detailed knowledge of the radioactive processing and disposal systems of the facility and the hazards associated therewith.

In special situations, such as facilities which produce and ship isotopes or irradiated experiments, the senior operator may need some knowledge of packaging and shipping regulations for radioactive materials, if the scope of his activities at the facility encompass such responsibilities.

J. SPECIFIC OPERATING CHARACTERISTICS

This category contains questions on specific operating characteristics of the reactor and auxiliary ystems, including nuclear, hydraulic, thermal, pneumatic, electrical and coolant chemistry. Questions regarding quantitative as well as qualitative explanations of causes, limitations, effects and consequence of changes are included.

The category includes questions on the understanding and use of curves depicting reactor behavior which may be beyond the scope of knowledge needed by operators for routine operation. These may include, as applicable, differential and integral control rod worth curves (single or group), period vs. reactivity curves, temperature and power coefficient curves, and poison (Xenon, Samarium, Boron, etc.,) worth curves. The curves will be given with the examination questions. Whenever possible, actual curves of the facility will be utilized; otherwise, applicable sample illustrative curves will be prepared.

K. FUEL HANDLING AND CORE PARAMETERS

This category contains questions regarding fuel, fuel handling and core loading, including procedures and limitations concerning core loading and alteration, fuel transfer and storage, and detection and prevention of criticality. Questions relating to fuel element characteristics and limitations include consideration of reactivity worths, fuel burnup, thermal hot spots, cladding rupture detection, effects of boiling and control rod programming. The applicant should be able to determine the reactivity status of the reactor based on the facility's parameters and coefficients.

Curves and mathematical expressions may be utilized to the extent described in category J. Knowledge of special equipment, procedures and personnel requirements regarding fuel handling and core loading is expected.

L. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS

This category contains questions on administrative, procedural and regulatory items which affect operation of the facility. Included are questions on design and operating considerations and limitations as specified in the facility license, including technical specifications, the procedures required to obtain authority for design changes, the procedures regarding formation and

approval of operating procedures, the authority to approve deviations from operating procedures on either a permanent or temporary basis, and emergency situations as they affect the entire plant's operation or security. Questions concerning the technical specifications will require a thorough knowledge of what items are addressed in the specifications including how to comply with the requirements. The operating limits and the bases for the specifications should be generally understood. The exact details, numbers and surveillance requirements contained therein are not expected to be memorized. Questions may also cover the requirements for certain personnel to be present at certain times, the types of records that must be maintained, knowledge of the facility radiological emergency plan, applicable positions of the Quality Assurance for Operations program, and pertinent provisions of 10 CFR Part 50 and Part 55.

VI. RELATIONSHIP OF CATEGORIES

The nature of the facility for which a license is sought influences the knowledge and duties of an operator at that facility, and correspondingly affects the relative emphasis between categories in the examination. For example, at a large facility, such as a power reactor, there are a large number of design features, there are several safety and emergency systems and there is a comprehensive system of operating procedures and administrative controls. Consequently, these aspects may be emphasized in the examination more than they are in the examination for a small facility. At a large plant, there is usually a Health Physics group providing continuous coverage; therefore, this aspect may receive less emphasis in the examination than it would receive in an examination at a small facility.

Conversely, at a small facility, such as a research or critical facility, operational safety is usually more dependent on the operators knowledge of reactor behavior and his understanding of instrumentation than it is at a large facility. Therefore, these items may receive proportionately higher emphasis in the examination.

VII. OPERATING TEST ADMINISTRATIVE CONSIDERATIONS

Certain administrative premises remain in force during the operating test. Among these are the following:

A. Neither the administering of an examination nor the presence of the examiner alters the normal chain of command as set forth in the facility's administrative controls, the facility license or the regulations. During his manipulation of controls (refer to Appendix F) in an examination, the applicant should obtain all permissions and make all reports that would normally be required of an operator at that facility. All directions to the applicant will emanate from the shift supervisor or his alternate in accordance with the facility administrative procedures. The examiner will only interrogate and make requests for actions.

B. Throughout the examination, all conditions of the license and provisions of the facility procedures should be observed. No request by the examiner for the applicant to perform an illegal or unsafe act will ever intentionally be made. If a requested action appears to be of this type, the applicant, licensed operator, senior operator, or his supervisor should so indicate immediately.

C. In order to ensure the integrity of an examination, reduce distractions and lessen extrinsic influences on the applicant, the number of persons present should be minimized. Other than the applicant and NRC personnel, only the minimum number of facility staff that are necessary for the examination to be performed should be present. This consideration is not intended to detract from the right and duty of the facility management to ensure safe operation at all times.

D. The applicant may refer to and use any applicable material normally provided to the operating staff. These include system operating procedures, alarm, off-normal and emergency procedure followup actions, technical specifications, operating curves, system prints and other relevant material.

VIII. SCOPE OF OPERATING TEST FOR OPERATORS

The scope of the operating test is set forth in Section 55.23 of Part 55 and is administered to the extent applicable at the facility. The applicant is expected to demonstrate sufficient understanding of the items listed in this section.

The following items will normally be included in all operating-oral examinations. (Refer to Appendix G for the Examination Report and Operating and Oral Examination Checklist used by examiners during the conduct of these examinations.) However, the specific characteristics and operating conditions of the facility for which a license is sought will be taken into consideration to ensure that the examination accomplishes its objectives. Therefore, for each examination, additions, deletions or modifications to the following list may be appropriate:

1. The applicant will be asked to perform a prestartup check on the reactor and any other checks (e.g., daily, recovery from scram) that all licensed operators would normally perform.

2. The applicant will be asked to identify selected nuclear instrumentation channel components and be able to describe the basic principles of operation for the nuclear detectors associated with each channel.

3. The applicant will be asked to start up the reactor from a substantially subcritical condition and raise power to a preselected value requested by the examiner and authorized by the reactor supervisor which is sufficient to utilize selected nuclear instrumentation channels and to introduce effects on reactivity (e.g., temperature increase, void formation) as may be appropriate.*

4. The applicant will be asked to describe his actions and responses to selected alarm and annunciator signals and indicate the probable causes and significance thereof. The applicant should show a high degree of familiarity with procedures of this nature and should distinguish between actions or checks which he must take immediately and those actions which are logical followups depending on the circumstances.

*If the alternative program, described in Appendix F, has been completed, this phase of operation will only be discussed with the applicant. No actual manipulations will be performed.

5. The applicant will be asked to predict the approximate readings of pertinent instrumentation for the conditions at which he will operate and to verify that his predictions are accurate.*

6. The applicant will be asked to describe the response of the system to control changes and verify that his description is correct. Normally, the applicant will be asked to make one or more changes of power level on a period requested by the examiner and authorized by the reactor supervisor.*

7. The applicant will be asked to demonstrate familiarity with auxiliary and experimental systems at the facility, if applicable, and particularly to indicate the interrelationships and interconnections between them and the reactor or reactor control system.

8. The applicant may be asked to perform such standard calculations (e.g., estimated critical prediction, period, etc.) as are consistent with an operator's responsibility at the facility.

9. The applicant will be asked to align and start, or describe the procedure for aligning and starting, several of the pertinent auxiliary and emergency systems.

10. The applicant will be asked to describe the operation and pertinent design and construction features of selected reactor and auxiliary systems and to display substantial familiarity with the overall facility, including the ability to locate and identify significant components and instrumentation.

11. The applicant will be asked to demonstrate the use of and interpret the readings of the fixed monitoring systems and personnel and portable monitoring equipment that is usually available to the operator at the facility.

12. The applicant will be asked to demonstrate his actions and describe his responsibilities in the event of emergencies that may occur. The applicant will be asked to describe symptoms of off-normal and emergency conditions and will be asked to demonstrate familiarity with resultant automatic and immediate actions without reference to control room reference material. This will include location of controls and indications which require manipulation or verification. A high degree of familiarity with the emergency procedures is required, and the applicant should be able to distinguish between those actions he must take immediately as an operator, those which are followup actions for operators and those which other personnel at the facility must follow.

13. The applicant should observe all rules and procedures regarding radiation safety and equipment, radiation work permits and permissions required, and demonstrate a logical safe approach to questions involving radiological safety, including hypothesized situations.

*If the alternative program, described in Appendix F, has been completed, this phase of operation will only be discussed with the applicant. No actual manipulations will be performed.

14. The applicant should demonstrate familiarity with and follow all operating procedures and standards of the facility, including his responsibility for careful and safe operation of the facility, notification of supervisory and other facility personnel and the obtaining of permission when required.

15. The applicant should demonstrate a familiarity with the contents of the technical specifications, including operating restrictions and limitations, to the extent they are directly applicable to an operator.

IX. SCOPE OF OPERATING TEST FOR SENIOR OPERATORS

The scope of the operating examination for the applicants for a senior operator's license will be generally the same as that outlined above for operators. The principal differences are:

1. In most areas of discussion, a senior operator candidate should demonstrate a higher degree of competence, knowledge and understanding than that required of an operator.

2. A senior operator candidate should display wider and more thorough knowledge of administrative controls, facility license, technical specifications and provisions of applicable regulations.

3. A senior operator candidate should demonstrate more breadth of knowledge of the facility, since he should be familiar with some functions or areas of the facility (e.g., disposition of stored radioactive waste), which are beyond the scope of knowledge expected of an operator.

X. WAIVER OF EXAMINATION AND TEST REQUIREMENTS

On application, the Commission may waive any or all of the requirements for a written examination and operating test if it finds that:

1. The applicant has had extensive actual operating experience at a comparable facility within two years prior to the date of application. A comparable reactor would normally be one where the class and operating characteristics of the reactor are similar to those of the reactor for which the applicant seeks a license (e.g., pressurized water power reactor, pool research reactor, etc.). Applicable experience at a comparable reactor is not limited to licensed facilities. The granting of partial or complete waiver of examinations will depend upon both the extent and responsibilities of the applicant's prior operating experience, the characteristics and complexity of the facility, the similarity between the reactors being compared, and the applicant's past performance record as indicated by information contained in his docket file.

2. The applicant has discharged his responsibilities competently and safely and is capable of continuing to do so. The Commission may accept as proof of the applicant's past performance a certification of an authorized representative of the facility licensee by which

the applicant was previously employed. The certification shall contain a description of the applicant's operating experience, including the number of startups and shutdowns and approximate total operating hours during which the applicant operated the controls of the facility, the duties performed, and the extent of his responsibility.

3. The applicant has learned the operating procedure for and is qualified to operate competently and safely the facility designated in his application. The Commission may accept as proof of the applicant's qualifications a certification of an authorized representative of the facility licensee (refer to item II.6) where the applicant's services will be utilized.

XI. ADMINISTRATION OF OPERATING TEST PRIOR TO INITIAL CRITICALITY

Except for the provisions of 10 CFR Part 55.9 concerning training, the regulations permit manipulation of the controls of a facility only by persons holding operator's licenses at that facility. This provision includes the period covering initial loading and startup. Therefore, it is necessary for examinations to be given and licenses granted prior to initial criticality. Obviously, an actual startup demonstration is not possible and can only be simulated. These examinations are termed "cold" examinations as opposed to examinations given at an operating reactor, termed "hot" examinations.

The requirements regarding eligibility for cold examinations are more stringent and the examinations more rigorous for two reasons.

1. Since the applicant can only simulate operation of the reactor, and the instrument responses and behavior characteristics are not reflected by this method, a more thorough and comprehensive examination is required for an examiner to make determinations regarding competence and knowledge.

2. The initial loading and startup of the reactor is in a period when the design parameters and expected operating characteristics have not been fully confirmed. Therefore, in this period, a relatively higher degree of uncertainty exists and the operators should be commensurately better qualified.

10 CFR Part 55.25 provides that the Commission may administer cold examinations, if a written request by an authorized representative of the facility licensee is sufficient for the Commission to determine that:

1. There is an immediate need for the services of the operator or senior operator license applicant.

   NOTE: The size of the facility and its operating schedule will usually govern the number of licenses necessary for proper coverage of reactor operation. The number of persons for whom licensing is planned prior to criticality should be sufficient to assure that applicable

technical specification conditions with respect to the number of licensed operators or shift crews can be met from the time of initial fuel loading, with due allowance given for examination contingencies.

2. The applicant has had extensive actual operating experience at a comparable reactor.

Under present NRC procedures for determining the eligibility of an applicant to take a cold examination, the applicant is considered to have had extensive operating experience at a comparable facility if one of the following four conditions have been met:

a. Holding or having held an operator's or senior operator's license at a comparable licensed reactor facility. To date, NRC has considered any light water power reactor as comparable to any other light water power reactor. However, it is highly desirable that previously licensed individuals participate in a short course utilizing a nuclear power plant simulator similar to the facility for which the applicant will be seeking a license. This training should take place as close to fuel loading as practicable.

b. Determination of such experience as indicated in a. above at a comparable reactor facility not subject to NRC licensing (e.g., reactor facilities operated by the military services or owned by the Energy Research and Development Administration (ERDA)).

A certification from the facility management of the applicant's experience may be acceptable in making this determination. The certification shall contain a description of the applicant's operating experience, including the number of startups, shutdowns, and approximate total hours during which the applicant operated the facility controls, the duties performed and the extent of his responsibilities. It is desirable that the previous duties and responsibilities be similar to those he will assume at the facility for which he seeks a license.

In addition, it is highly desirable that ex-military personnel participate in a short course utilizing a nuclear power plant simulator similar to the facility for which the applicant will be seeking a license.

Applicants who have been certified at ERDA-owned reactors and lack power plant experience are required to attend an appropriate nuclear power simulator course or participate in the day-to-day operations of a plant similar to the one for which he seeks a license for a period of two months.

c. The applicant has passed an NRC-administered written examination and operating test at a comparable licensed reactor facility after completing an NRC approved training program. Normally, the Commission issues a certification stating that the individual has met the requirements of an operator as set forth in 10 CFR Part 55 in lieu of a license.

d. Certification of satisfactory completion of an NRC-approved training program which utilizes a nuclear power plant simulator as part of the program.

Extensive actual operating experience, as detailed above, must be obtained within a reasonable time period prior to the administration of the "cold" examination. Normally, a period of 24 months prior to the "cold" examination is considered a reasonable time period.

3. The applicant has a thorough knowledge of the reactor control system, instrumentation, and operating procedures under normal, abnormal and emergency conditions.

4. The reactor control mechanism and instrumentation are in such condition as determined by the Commission to permit effective administration of a simulated operating test.

XII. ISSUANCE OF LICENSES

Section 55.11 of 10 CFR Part 55 lists three requirements to be fulfilled prior to issuance of a license. These may be briefly summarized as:

1. A satisfactory medical determination.

2. Successful completion of the written and operating tests.

3. The applicant's services will be utilized.

After determining that an application meets the above requirements, a license will be issued. Every license contains certain conditions, as specified in Section 55.31 of 10 CFR Part 55, and may also contain such other conditions and limitations as the Commission deems appropriate and necessary.

XIII. EXPIRATION OF LICENSES

Each license shall expire two years after the date of issuance unless earlier action on the license has been taken by the Commission. A licensee may apply for renewal of his license at a facility if there is a continuing need for his licensed services at that facility. [Note: the date of issuance is the effective date on the license. It is not the amended date.]

XIV. TIMELY RENEWAL APPLICATIONS

In any case in which a licensee has filed an application in proper form for renewal or for a new license not less than thirty days prior to the expiration of his existing license, the existing license shall not expire until the application for renewal or for a new license has been finally determined by the Commission.

XV. CONTENT OF RENEWAL APPLICATIONS

An application for renewal of a license should contain all of the information set forth in Section 55.33 of 10 CFR Part 55, as follows:

1. The full name, citizenship, address and present employment of the applicant (reference Section II.1 above). The information regarding education and past experience (reference section II.2 above) need not be repeated, but any educational achievements attained during the license period should be listed in order to update the information previously supplied.

2. The serial number of the license for which renewal is sought.

3. The experience of the applicant under his existing license, including the approximate number of hours during which he has operated the facility. Quantitative information other than hours should also be included if it more clearly represents the applicant's operating experience. For example, at a critical facility, the number of startups performed is often a more significant indication than the number of hours of operation.

   For a senior operator applicant, the extent of his activity in directing operations should be reported, as well as his personal operating experience.

   The report of experience should include such other aspects as may be pertinent to the continued competence of the applicant even though the activity itself is not a licensed one. An example of such an activity is the preparation of technical specifications.

4. A description of the requalification program that has been successfully completed during the effective term of his license. The Commission requires that each person holding a facility operating license issued pursuant to 10 CFR Part 50 administer continuous requalification programs for licensed operators and senior operators. The requalification program must be approved by the Commission. Records must be maintained at each facility that indicate the extent of participation of each licensed operator and senior operator, including the facility management's evaluation of the individuals. Consequently, the renewal application need only reference successful participation in the approved requalification program.

5. Evidence that the licensee has discharged his license responsibilities safely and competently. The Commission may accept as evidence of this a certificate of an authorized representative of the facility licensee by which the licensee has been employed.

6. A report of medical examination by a licensed medical practitioner in the form prescribed in Section 55.60 of 10 CFR Part 55. This form, NRC-396, is attached as Appendix D to this guide.

XVI. RENEWAL OF LICENSES

Section 55.33(c) of 10 CFR Part 55 lists three requirements to be fulfulled prior to renewal of a license. These are:

1. A satisfactory medical determination.

2. A finding of extensive and active engagement as an operator or senior operator under his existing license, competent prior operation and the capability of continuing to do so, and successful participation in the facility requalification program or successful completion of prescribed reexamination. To be considered actively engaged under his license, the individual's duties must be performed at the facility, normally on a day-to-day basis.

3. Continued need for a license.

XVII. DENIAL OF APPLICATIONS

An applicant who fails to meet one or more of the requirements for approval of an application will be issued a denial of his application by the Commission. The letter of denial will contain the reasons for the denial and a reference to the regulations regarding reapplication and the opportunity for a hearing on the denial. If the cause of the denial is failure to pass the written examination, operating test or both, the provisions for re-application which apply are set forth in Section 55.12 of 10 CFR Part 55. If the application is made for a senior operator's license, and the applicant fails to pass the senior written examination or is deficient in the operating test at the senior operator level, the applicant may be issued an operator's license, provided he passed the operator portion of the written examination and the operating test at the operator level. If the cause of the denial is medical in nature, the provisions for further submission under the original application are set forth in Section 55.10(c) of 10 CFR Part 55.

"Cold" examination applicants are normally administered the written examination several weeks in advance of the operating test. In these cases, those who fail the operator written examination will be issued a denial letter and no operating test will be administered. Those who fail the senior written examination but pass the operator written examination, may be administered an operating test at the senior level. See Item IX above. However, only an operator's license will be issued. The Commission may, on request, issue a denial letter to those who failed only the senior written examination at the time of the failure rather than at the completion of the operating test.

XVIII. NOTIFICATION OF DISABILITY SUBSEQUENT TO LICENSING

During the effective period of a license, the licensee shall, within fifteen days after its occurrence, notify the Commission of any disability referred to in 10 CFR Part 55.11(a)(1) which occurs after the submission of his medical examination form. Automobile or industrial accidents, injuries resulting from sporting events or any accident which reduces dexterity or mental alterness should be considered as grounds to notify the Commission and temporarily remove the licensee from licensed duties.

XIX. MULTI-UNIT FACILITIES

Pursuant to 10 CFR Part 55.31(b) the license is limited to the facility for which it is issued. For sites which have more than one unit located thereon, each unit is considered a separate facility. Consequently, individuals licensed on one unit must submit applications for the subsequent units as appropriate. Consideration will be given to requests for waiver of examination pursuant to Item X above.

UNITED STATES NUCLEAR REGULATORY COMMISSION
RULES and REGULATIONS
TITLE 10, CHAPTER 1, CODE OF FEDERAL REGULATIONS—ENERGY

PART 55

OPERATORS' LICENSES

GENERAL PROVISIONS

AUTHORITY: The provisions of this Part 55 issued under secs. 107, 161, 68 Stat. 939, 948; 42 U.S.C. 2137, 2201. For the purposes of sec. 223, 68 Stat. 958, as amended; 42 U.S.C. 2273, § 55.3 issued under sec. 161i., 68 Stat. 949; 42 U.S.C. 2201 (i). Sec. 55.40 issued under secs. 186, 187, 68 Stat. 955; 42 U.S.C. 2236, 2237. Secs. 202, 206, Publ. L. 93-438, 88 Stat. 1244, 1246; 42 U.S.C. 5842, 5846.

GENERAL PROVISIONS

§ 55.1 Purpose.

28 FR 3197

The regulations in this part establish procedures and criteria for the issuance of licenses to operators, including senior operators, of facilities licensed pursuant to the Atomic Energy Act of 1954, as amended, or section 202 of the Energy Reorganization Act of 1974 and Part 50 of this chapter; and provide for the terms and conditions upon which the Commission will issue these licenses.

§ 55.2 Scope.

40 FR 8774

The regulations contained in this part apply to any individual who manipulates the controls of any facility licensed pursuant to Part 50 of this chapter and to any individual designated by a facility licensee to be responsible for directing the licensed activities of licensed operators.

§ 55.3 License requirements.

28 FR 3197

(a) No person may perform the function of an operator as defined in this part except as authorized by a license issued by the Commission;

(b) No person may perform the function of a senior operator as defined in this part except as authorized by a license issued by the Commission.

§ 55.4 Definitions.

As used in this part:

(a) "Act" means the Atomic Energy Act of 1954 including any amendments thereto.

(b) "Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.

(c) "Facility" means any production facility or utilization facility as defined in Part 50 of this chapter.

(d) "Operator" is any individual who manipulates a control of a facility. An individual is deemed to manipulate a control if he directs another to manipulate a control.

28 FR 3197

(e) "Senior operator" is any individual designated by a facility licensee under Part 50 of this chapter to direct the licensed activities of licensed operators.

(f) "Controls" when used with respect to a nuclear reactor means apparatus and mechanisms the manipulation of which directly affect 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.

(g) "United States" when used in a geographical sense, includes all territories and possessions of the United States, the Canal Zone and Puerto Rico.

§ 55.5 Communications.

40 FR 8774

Except where otherwise specified, all communications and reports concerning the regulations in this part, and applications filed under them should be addressed to the Director of Nuclear Reactor Regulation or the Director of Nuclear Material Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street, N.W., Washington, D.C. or at 7920 Nor-

folk Avenue, Bethesda, Md.

§ 55.6 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.

EXEMPTIONS

§ 55.7 Specific exemptions.

The Commission may, upon application by an interested person, or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property and are otherwise in the public interest.

28 FR 3197

§ 55.8 Additional requirements.

The Commission may, by rule, regulation, or order, impose upon any licensee such requirements in addition to those established in the regulations in this part, as it deems appropriate or necessary to protect health and to minimize danger to life or property.

§ 55.9 Exemptions from license.

Nothing in this part shall be deemed to require a license for:

(a) An individual who manipulates the controls of a research or training reactor as part of his training as a student in a nuclear engineering course under the direction and in the presence of a licensed operator or senior operator;

(b) An individual who manipulates the controls of a facility as a part of his training to qualify for an operator license under this part under the direction and in the presence of a licensed operator or senior operator.

LICENSE APPLICATIONS

§ 55.10 Contents of applications.

40 FR 8774

(a) Applications for licenses should be filed in triplicate, except for the report of medical examination, with the Director of Nuclear Reactor Regulation or the Director of Nuclear Material Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street NW., Washington, D.C., or 7920 Norfolk Avenue, Bethesda, Md.

31 FR 4668

Each application for a license shall contain the following information:

(1) The full name, citizenship, age, address and present employment of the applicant;

(2) The education and pertinent experience of the applicant, including detailed information on the extent and nature of responsibility;

(3) Serial numbers of any operator and senior operator license issued by the Commission to the applicant and the expiration date of each;

(4) The specific facility for which the applicant seeks an operator or senior operator license;

(5) The written request of an authorized representative of the facility license that the operating test be administered to the applicant of the facility.

(6) Evidence that the applicant has learned to operate the controls in a competent and safe manner and has need for an operator or a senior operator license in the performance of his duties. The Commission may accept as proof of this a certification of an authorized representative of the facility licensee where the applicant's services will be utilized. This certification shall include details on courses of instruction administered by the facility licensee, number of course hours, number of hours of training and nature of training received at the facility, and for reactors, the startup and shutdown experience received.

28 FR 3197

(7) A report of a medical examination by a licensed medical practitioner, in one copy in the form prescribed in § 55.60.

(b) The Commission may at any time after the filing of the original application, and before the expiration of the license, require further information in order to enable it to determine whether the application should be granted or denied or whether a license should be revoked, modified or suspended.

(c) An applicant whose application has been denied because of his physical condition or general health may submit a further report of medical examination at any time as a supplement to his original application.

(d) Each application and statement shall contain complete and accurate disclosure as to all matters and things required to be disclosed. All applications and statements, other than the matters required by items 5, 6, and 7 of paragraph (a) of this section shall be signed by the applicant.

§ 55.11 Requirements for the approval of application.

An application for a license pursuant to the regulations in this part will be approved if the Commission finds that:

(a) The physical condition and the general health of the applicant are not such as might cause operational errors endangering public health and safety.

(1) Epilepsy, insanity, diabetes, hypertension, cardiac disease, fainting spells, defective hearing or vision or any other physical or mental condition which might cause impaired judgment or motor coordination may constitute sufficient cause for denial of an application.

(2) If an applicant's vision, hearing and general physical condition do not meet the minimum standards normally considered necessary, the Commission may approve the application and include conditions in the license to accommodate the physical defect. The Commission will consider the recommendations of the facility licensee or holder of an authorization and of the examining physician on Form NRC—396 in arriving at its decision.

28 FR 3197

(b) The applicant has passed a written examination and operating test as may be prescribed by the Commission to determine that he has learned to operate and, in the case of a senior operator, to operate and to direct the licensed activities of licensed operators in a competent and safe manner.

(c) The applicant's service as a licensed operator or senior operator will be utilized on the facility for which he seeks a license or on a similar facility within the United States.

§ 55.12 Re-applications.

(a) Any applicant whose application for a license has been denied because of failure to pass the written examination or operating test or both may file a new application for license two months after the date of denial. Any new application shall be accompanied by a statement signed by an authorized representative of the facility licensee by whom the applicant will be employed, stating in detail the extent of additional training which the applicant has received and certifying that he is ready for re-examination. An applicant may file a third application six months after the date of denial of his second application, and may file further successive applications two years after the date of denial of each prior application.

(b) An applicant who has passed either the written examination or operat-

ing test and failed the other may request in a new application that he be excused from re-examination on the examination or test which he has passed. The Commission may in its discretion grant the request if it determines that sufficient justification is presented under all the circumstances.

WRITTEN EXAMINATIONS AND OPERATING TESTS

§ 55.20 Scope of examinations.

The written examination and operating test for a license as an operator or a senior operator are designed to test the applicant's understanding of the facility design and his familiarity with the controls and operating procedures of the facility. The written examination is based in part on information in the final* safety analysis report, operating manuals, and license for the facility.

§ 55.21 Content of operator written examination.

The operator written examination, to the extent applicable t the facility, will include questions on:

(a) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, and criticality indications.

(b) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

(c) Mechanical design features of the reactor primary system.

(d) Auxiliary systems which affect the facility.

(e) General operating characteristics, including causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for them.

(f) Design, components and functions of reactivity control mechanisms and instrumentation.

(g) Design, components and functions of safety systems, including instrumentation, signals, interlocks, automatic and manual features.

(h) Components, capacity and functions of reserve and emergency systems.

(i) Shielding, isolation and containment design features, including access limitations.

(j) Standard and emergency operating procedures for the facility and plant.

(k) Purpose and operation of radiation monitoring system, including alarm and survey equipment.

*Amended 31 FR 12774.

(l) Radiological safety principles and procedures.

§ 55.22 Content of senior operator written examination.

The senior operator written examination to the extent applicable to the facility, will include questions on the items specified in § 55.21 and in addition on the following:

(a) Conditions and limitations in the facility license.

(b) Design and operating limitations in the technical specifications for the facility.

(c) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

(d) Radiation hazards which may arise during the performance of experiments, shielding alterations, maintenance activities and various contamination conditions.

(e) Reactor theory, including details of fission process, neutron multiplication, source effects, control rod effects, and criticality indications.

(f) Specific operating characteristics, including coolant chemistry and causes and effects of temperature, pressure and reactivity changes.

(g) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, determination of various internal and external effects on core reactivity.

(h) Fuel handling facilities and procedures.

(i) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

§ 55.23 Scope of operator and senior operator operating tests.

The operating tests administered to applicants for operator and senior operator licenses are generally similar in scope. The operating test, to the extent applicable to the facility requires the applicant to demonstrate and understanding of:

(a) Pre-start-up procedures for the facility, including associated plant equipment which could affect reactivity.

(b) Required manipulation of console controls to bring the facility from shutdown to designated power levels.

(c) The source and significance of annunciator signals and condition-indicating signals and remedial action responsive thereto.

(d) The instrumentation system and the source and significance of reactor instrument readings.

(e) The behavior characteristics of the facility.

(f) The control manipulation required to obtain desired operating results during normal, abnormal and emergency situations.

(g) The operation of the facility's heat removal systems, including primary coolant, emergency coolant, and decay heat removal systems, and the relation of the proper operation of these systems to the operation of the facility.

(h) The operation of the facility's auxiliary systems which could affect reactivity.

(i) The use and function of the facility's radiation monitoring systems, including fixed radiation monitors and alarms, portable survey instruments, and personnel monitoring equipment.

(j) The significance of radiation hazards, including permissible levels of radiation, levels in excess of those authorized and procedures to reduce excessive levels of radiation and to guard against personnel exposure.

(k) The emergency plan for the facility, including the operator's or senior operator's responsibility to decide whether the plan should be executed and the duties assigned under the plan.

(l) The necessity for a careful approach to the responsibility associated with the safe operation of the facility.

§ 55.24 Waiver of examination and test requirements.

On application, the Commission may waive any or all of the requirements for a written examination and operating test if it finds that the applicant:

(a) Has had extensive actual operating experience at a comparable facility within two years prior to the date of application.

(b) Has discharged his responsibilities competently and safely and is capable of continuing to do so. The Commission may accept as proof of the applicant's past performance a certification of an authorized representative of the facility licensee or holder of an authorization by which the applicant was previously employed. The certification shall contain a description of the applicant's operating experience, including an approximate number of hours the applicant operated the controls of the facility, the duties performed, and the extent of his responsibility.

(c) Has learned the operating procedure for and is qualified to operate competently and safely the facility designated in his application. The Commission may accept as proof of the applicant's qualifications a certification of an authorized representative of the facility licensee or holder of an authorization

where the applicant's services will be utilized.

§ 55.25 Administration of operating test prior to initial criticality.

The Commission may administer a simulated operating test to an applicant for a license to operate a reactor prior to its initial criticality if a written request by an authorized representative of the facility licensee is sufficient for the Commission to find that:

(a) There is an immediate need for the applicant's services.

(b) The applicant has had extensive actual operating experience at a comparable reactor.

(c) The applicant has a thorough knowledge of the reactor control system, instrumentation and operating procedures under normal, abnormal, and emergency conditions.

(d) The reactor control mechanism and instrumentation are in such condition as determined by the Commission to permit effective administration of a simulated operating test.

28 FR 3197

LICENSES

§ 55.30 Issuance of licenses.

On determining that an application meets the requirements of the Act and the regulations of the Commission, the Commission will issue a license in such form and containing such conditions and limitations as it deems appropriate and necessary.

§ 55.31 Conditions of the licenses.

Each license shall contain and is subject to the following conditions, whether stated in the license or not:

(a) Neither the license nor any right under the license shall be assigned or otherwise transfered.

(b) The license is limited to the facility for which it is issued.

(c) The license is limited to those controls of the facility specified in the license.

(d) The license is subject to, and the licensee shall observe, all applicable rules, regulations and orders of the Commission.

38 FR 22221

(e) If a licensee has not been actively performing the functions of an operator or senior operator for a period of four months or longer, he shall, prior to resuming activities licensed pursuant to this part, demonstrate to the Commission that the knowledge and understanding of facility operation and administration are satisfactory. The Commission may accept as evidence, a certification by an authorized representative of the facility licensee by which the licensee has been employed.

28 FR 3197

‡(f) Such other conditions as the Commission may impose to protect health or to minimize danger to life or property.

§ 55.32 Expiration.

Each operator and senior operator licensee shall expire two years after the date of issuance.

§ 55.33 Renewal of licenses.

(a) Application for renewal of a license shall be signed by the applicant and shall contain the following information:

(1) The full name, citizenship, address and present employment of the applicant;

(2) The serial number of the license for which renewal is sought;

(3) The experience of the applicant under his existing license, including the approximate number of hours during which he has operated the facility;

38 FR 22221

(4) A statement that during the effective term of his current license the applicant has satisfactorily completed the requalification program for the facility for which operator or senior operator license renewal is sought. In the case of an application for license renewal filed within two years after September 17, 1973, if the facility licensee has not implemented the requalification program requirements in time for the applicant to complete an approved requalification program before the effective term of his current license expires, the applicant shall submit a statement showing his current enrollment in an approved requalification program and describing those portions of the program which he had completed by the date of his application for license renewal.

28 FR 3197

‡(5) Evidence that the licensee has discharged his license responsibilities competently and safely. The Commission may accept as evidence of this a certificate of an authorized representative of the facility licensee or holder of an authorization by which the licensee has been employed;

‡(6) A report by a licensed medical practitioner in the form prescribed in § 55.60.

(b) In any case in which a licensee not less than thirty days prior to the expiration of his existing license has filed an application in proper form for renewal or for a new license, the existing license shall not expire until the application for renewal or for a new license has been finally determined by the Commission.

28 FR 3197

(c) The licensee will be renewed if the Commission finds that:

(1) The physical condition and the general health of the licensee continue to be such as not to cause operational errors which migh endanger public health and safety; and

38 FR 22221

(2) (i) The licensee has been actively and extensively engaged as an operator or as a senior operator under his existing license, has discharged his responsibilities competently and safely, and is capable of continuing to do so.

(ii) The licensee has completed a requalification program or is presently enrolled in a requalification program if the completion of the requalification program will occur after the expiration of his license as provided in subparagraph (a) (4) of this section.

(iii) If the requirements of paragraph (c) (2) (i) and (ii) of this section are not met, the Commission may require the applicant for renewal to take a written examination or an operating test or both.

(3) There is a continued need for a license to operate or direct operators at the facility designated in the application.

MODIFICATION AND REVOCATION OF LICENSES

§ 55.40 Modification and revocation of licenses.

(a) The terms and conditions of all licenses shall be subject to amendment, revision, or modification by reason of amendments to the Act, or by reason of rules, regulations or orders issued in accordance with the Act or any amendments thereto.

28 FR 3197

(b) Any license may be revoked, suspended or modified, in whole or in part, for any material false statement in the application or any statement of fact required under section 182 of the Act, or because of conditions revealed by such application 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, or for violation of, or failure to observe any of the terms and conditions of the Act, or the license, or of any rule, regulation or order of the Commission, or any conduct determined by the Commission to be a hazard to safe operation of the facility.

‡Redesignated 38 FR 22221

41 FR 16445

§ 55.41 Notification of disability.

The licensee shall notify the Director of Nuclear Reactor Regulation or the Director of Nuclear Material Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within fifteen (15) days after its occurrence of any disability referred to in § 55.11(a)(1) which occurs after the submission of his medical examination form.

ENFORCEMENT

40 FR 8774

§ 55.50 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, or Title II of the Energy Reorganization Act of 1974, or any regulation or order issued thereunder. A court order may be obtained for the payment of a civil penalty imposed pursuant to section 234 of the Act for violation of section 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Act, or section 206 of the Energy Reorganization Act of 1974, or any rule, regulation, or order issued thereunder, or any term, condition, or limitation of any license issued thereunder, or for any violation for which a license may be revoked under section 186 of the Act. Any person who willfully violates any provision of the Act or any regulation or order issued thereunder may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

CERTIFICATE OF MEDICAL EXAMINATION

28 FR 3197

§ 55.60 Examination form.

(a) An applicant shall complete and sign Form NRC—396, "Certificate of Medical Examination."

(b) The examining physician shall complete and sign Form NRC—396 and shall mail the completed form to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555.

NOTE: Copies of Form NRC—396 may be obtained by writing to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555.

§ 55.61 [Deleted 40 FR 8774.]

38 FR 22221

APPENDIX A

REQUALIFICATION PROGRAMS FOR LICENSED OPERATORS OF PRODUCTION AND UTILIZATION FACILITIES

Introduction

Section 50.54 of 10 CFR Part 50 requires that individuals who manipulate controls of production and utilization facilities be licensed as operators by the Commission and that individuals who direct the licensed activities of licensed operators be licensed as senior operators in accordance with 10 CFR Part 55. Section 55.33 of 10 CFR Part 55 requires that each licensed individual demonstrate his continued competence every two years in order for his license to be renewed. Competence may be demonstrated, in lieu of reexamination, by satisfactory completion of a requalification program which has been reviewed and approved by the Commission.

Periodic requalification for all operators and senior operators of production and utilization facilities is necessary for the personnel to maintain competence, particularly to respond to abnormal and emergency situations. The complexity of design and operating modes of production and utilization facilities require that ongoing comprehensive requalification programs be conducted for all licensed operators and senior operators as a matter of sound principle and practice.

Licensed operators and senior operators of production and utilization facilities who have been actively and extensively engaged as operators or as senior operators shall participate in requalification programs meeting the requirements of this Appendix. Individuals who maintain operator or senior operator licenses for the purpose of providing backup capability to the operating staff shall participate in the requalification programs except to the extent that their normal duties preclude the need for specific retraining in particular areas. Licensed operators or senior operators whose licenses are conditioned to permit manipulation of specific controls only shall participate in those portions of the requalification program appropriate to the duties they perform.

The requalification program requirements involving manipulation of controls may be performed on the facility for which the operator is licensed. However, the use of a simulator as specified in Paragraphs 3e and 4d of this appendix is permissible and such use is encouraged.

Requalification Program Requirements

1. *Schedule.* The requalification program shall be conducted for a continuous period not to exceed two years, and upon conclusion shall be promptly followed, pursuant to a continuous schedule, by successive requalification programs.

2. *Lectures.* The requalification program shall include preplanned lectures on a regular and continuing basis throughout the license period in those areas where annual operator and senior operator written examinations indicate that emphasis in scope and depth of coverage is needed in the following subjects:

a. Theory and principles of operation.
b. General and specific plant operating characteristics.
c. Plant instrumentation and control systems.
d. Plant protection systems.
e. Engineered safety systems.
f. Normal, abnormal, and emergency operating procedures.
g. Radiation control and safety.
h. Technical specifications.
i. Applicable portions of Title 10, Chapter I, Code of Federal Regulations.

Other training techniques including films, videotapes and other effective training aids may also be used.

Individual study on the part of each operator shall be encouraged. However, a requalification program based solely upon the use of films, videotapes and/or individual study is not an acceptable substitute for a lecture series.

3. *On-the-job training.* The requalification program shall include on-the-job training so that:

a. Each licensed operator of a production or utilization facility manipulates the plant controls and each licensed senior operator either manipulates the controls or directs the activities of individuals during plant control manipulations during the term of their licenses. For reactor operators and senior operators, these manipulations shall consist of at least 10 reactivity control manipulations in any combination of reactor startups, reactor shutdowns or other control manipulations which demonstrate skill and/or familiarity with reactivity control systems.

b. Each licensed operator and senior operator has demonstrated satisfactory understanding of the operation of all apparatus and mechanisms and knows the operating procedures in each area for which he is licensed.

c. Each licensed operator and senior operator is cognizant of facility design changes, procedure changes, and facility license changes.

d. Each licensed operator and senior operator reviews the contents of all abnormal and emergency procedures on a regularly scheduled basis.

e. A simulator may be used in meeting the requirements of paragraphs 3a* and 3b* if the simulator reproduces the general operating characteristics of the facility involved, and the arrangement of the instrumentation and controls of the simulator is similar to that of the facility involved.

4. *Evaluation.* The requalification program shall include:

a. Annual written examinations which determine areas in which retraining is needed to upgrade licensed operator and senior operator knowledge.

b. Written examinations which determine licensed operators' and senior operators' knowledge of subjects covered in the requalification program and provide a basis for evaluating their knowledge of abnormal and emergency procedures.

c. Systematic observation and evaluation of the performance and competency of licensed operators and senior operators by supervisors and/or training staff members including evaluation of actions taken or to be taken during actual or simulated abnormal and emergency conditions.

d. Simulation of emergency or abnormal conditions that may be accomplished by using the control panel of the facility involved or by using a simulator. Where the control panel of the facility is used for simulation, the actions taken or to be taken for the emergency or abnormal condition shall be discussed; actual manipulation of the plant controls is not required. If a simulator is used in meeting the requirements of paragraph 4c, the simulator shall accurately reproduce the operating characteristics of the facility involved and the arrangement of the instrumentation and controls of the simulator shall closely parallel that of the facility involved.

e. Provisions for each licensed operator and senior operator to participate in an accelerated requalification program where performance evaluations conducted pursuant to paragraphs 4a through 4d clearly indicate the need.

5. *Records.* Records of the requalification program shall be maintained to document each licensed operator's and senior operator's participation in the requalification program. The records shall contain copies of written examinations administered, the answers given by* the licensee, results of evaluations and documentation of any additional training administered in areas in which an operator or senior operator has exhibited deficiencies.

6. *Alternative training programs.* The requirements of this appendix may be met by requalification programs conducted by persons other than the facility licensee if such requalification programs are similar to the program described in paragraphs 1 through 5, and the alternative program has been ap-

*Amended 38 FR 26354

38 FR 2222

proved by the Commission.

7. *Applicability to research and test reactors and non-reactor facilities.* To accommodate specialized modes of operation and differences in control, equipment, and operator skills and knowledge, the requalification program for each licensed operator and senior operator of a research or test reactor or of a non-reactor facility shall conform generally but need not be identical to the requalification program outlined in paragraphs 1 through 6 of this appendix. However, significant deviations from the requirements of this appendix shall be permitted only if supported by written justification and approved by the Commission.

UNITED STATES NUCLEAR REGULATORY COMMISSION

RULES and REGULATIONS

TITLE 10, CHAPTER 1, CODE OF FEDERAL REGULATIONS—ENERGY

PART 50 LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

AUTHORITY: Secs. 103, 104, 161, 182, 183, 68 Stat. 936, 937, 948, 953, 954, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 (42 U.S.C., 5842, 5846), unless otherwise noted. Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended; 42 U.S.C. 2234. Sections 50.100–50.102 issued under sec. 186, 68 Stat. 955; 42 U.S.C. 2236. For the purposes of sec. 223, 68 Stat. 958, as amended; 42 U.S.C. 2273, § 50.54 (i) issued under sec. 1611, 68 Stat. 949; 42 U.S.C. 2201(i), and §§ 50.70–50.71 issued under sec. 161o, 68 Stat. 950, as amended; 42 U.S.C. 2201(o) and the Laws referred to in Appendices.

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 Titles II of the Energy Reorganization Act of 1974 (88 Stat. 1242) to provide for the licensing of production and utilization facilities.

§ 50.2 *Definitions.* As used in this part,

(a) "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

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(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

26 FR 4989

(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 unranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material.

39 FR 4871

(b) "Utilization facility" means any nuclear reactor other than one designed or used primarily for the formation of plutonium or U-233.

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.

(c) "Act" means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.

(d) "Agreement for cooperation" means any agreement with another nation or regional defense organization, authorized or permitted by sections 54, 57, 64, 82, 103, 104, or 144 of the act, and made pursuant to section 123 of the act.

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(e) "Atomic energy" means all forms of energy released in the course of nuclear fission or nuclear transformation.

(f) "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.

(g) "Byproduct 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.

(h) "Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.

(i) "Common defense and security" means the common defense and security of the United States.

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(j) "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.

(k) "Nuclear reactor" means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction.

(l) "Person" means (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission or the Administration, except that the Administration 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,[7] 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.

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(m) "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.

(n) "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.

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[7] The Administration 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.

(o) "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.

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*(p) "Source material" means source material as defined in subsection 11z. of the Act and in the regulations contained in Part 40 of this chapter.

**(q) "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.

(r) "Testing facility" means a nuclear reactor which is of a type described in § 50.21(c) 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.

25 FR 1072

*(s) "United States," when used in a geographical sense, includes all Territories and possessions of the United States, the Canal Zone, and Puerto Rico.

21 FR 355

(t) "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.

28 FR 3196

(u) "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.

33 FR 18610

(v) "Reactor coolant pressure boundary" means all those pressure-containing

*Redesignated 25 FR 1072.

**Corrected 31 FR 15145.

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 system 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.

36 FR 11424

(w) "Administration" means the Energy Research and Development Administration or its duly authorized representatives.

40 FR 8774

§ 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.

§ 50.4 Communications.

Except where otherwise specified, all communications and reports concerning the regulations in this part should be addressed to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, or may be delivered in person at the Commission's offices at 1717 H Street, NW., Washington, D.C. or at 7920 Norfolk Avenue, Bethesda, Maryland.

40 FR 8774

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, use, import, or export any production or utilization facility except as authorized by a license issued by the Commission.

25 FR 8712

(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;

33 FR 2381

(2) Procurement or manufacture of components of the facility;

(3) Construction of non-nuclear facilities (such as turbo-generators 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 104 a. or section 104 c. 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.)

25 FR 8712

This paragraph does not apply to production or utilization facilities subject to paragraph (c) of this section.

37 FR 5745

(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 §51.5(a)** 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:

39 FR 14506

(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.)

37 FR 5745

(d) (1) Each person subject to the provisions of paragraph (c) of this section, who is, on (effective date of these amendments), 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 subparagraph (2) of this paragraph (d), 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 subparagraph (2) of this paragraph (d).

(2) Upon submission of a statement of reasons pursuant to subparagraph (1) of this paragraph (d), the Commission may authorize the continued conduct of activities permitted by paragraph (b) of this section in effect prior to (effective date of these amendments), 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.

37 FR 5745

(e)(1) The Director of Nuclear Reactor Regulation may authorize an applicant for a construction permit for a nuclear power reactor subject to the provisions of §51.5(a)** of this chapter 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 are not subject to the provisions of Appendix B. No such authorization shall be granted unless the staff has completed a final environmental impact statement on the issuance of the construction permit as required by 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 §51.52(b) and (c)** of this chapter to be made prior to issuance of the construction permit for the facility, and (ii) has determined that, based upon the

39 FR 14506

*Effective date of these amendments.
**Amended 39 FR 26279.

available information and review to date, there is reasonable assurance that the proposed site is a suitable location for a nuclear power 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 nuclear power reactor subject to the provisions of §51.5(a)** of this chapter 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 are subject to the provisions of Appendix B.

(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.

39 FR 14506

§ 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[8] are involved,

(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 Administration under a prime contract for:

(A) The performance of work for the Administration 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 Administration 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 Administration 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 Administration 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 Administration in a United States government-owned vehicle or vessel: Provided, that such activities are conducted by a prime contractor of the Administration under a prime contract with the Administration.

(ii) The construction or operation of a production or utilization facility by a prime contractor or subcontractor of the Commission or the Administration under his 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.

(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 such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

37 FR 5745

(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.

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.

40 FR 8774

§ 50.13 Attacks and destructive acts by enemies of the United States; and defense activities.

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.

32 FR 13445

CLASSIFICATION AND DESCRIPTION OF LICENSES

§ 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.

21 FR 355

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, use, import, or export under the terms of an agreement for cooperation:

(a) A utilization facility for use in medical therapy; or

(b) (1) A production or utilization facility the construction or operation of which was licensed pursuant to subsection 104b of the Act prior to Decem-

[8] The Administration 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.

** Amended 39 FR 26279.

35 FR 19655

ber 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.

38 FR 11445

(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.

§ 50.22 Class 103 licenses; for commercial and industrial facilities.

(a) 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, use, import, or export under the terms of an agreement for cooperation, 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.

35 FR 19655

§ 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. 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]

31 FR 4668

APPLICATIONS FOR LICENSES, FORM, CONTENTS, INELIGIBILITY OF CERTAIN APPLICANTS

40 FR 8774

§ 50.30 Filing of applications for licenses; oath or affirmation.

(a) *Place of filing.* Each application for a license, including whenever appropriate a construction permit, or amendment thereof, should be filed, for a nuclear reactor, testing facility or other utilization facility, with the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Each application for a license, including where appropriate, a construction permit or amendment thereof, for a fuel reprocessing plant, should be filed with the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street, NW., Washington, D.C. or at 7920 Norfolk Avenue, Bethesda, Maryland.

23 FR 3115

(b) *Oath or affirmation.* Each application for a license, including whenever appropriate a construction permit or amendment thereof, should be executed in three signed originals by the applicant or duly authorized officer thereof under oath or affirmation.

34 FR 6036

(c) *Number of copies of applications.* (1) Each filing of an application for a license to construct and operate a production or utilization facility (including amendments to such applications) should include, in addition to the three signed originals, the following number of copies:

(i) For an application for a license for a facility described in § 50.21(b) or § 50.22, or a testing facility: 25 copies of that portion of the application containing the information required by § 50.33 (general information) and 70 copies of that portion of the application containing the information required by § 50.34 (safety analysis report);

(ii) For an application for an amendment to a license for a facility described in § 50.21(b) or § 50.22, or a testing facility: 19 copies of that portion of the application containing the information required by § 50.33 (general information) and 40 copies of that portion of the application containing the information required by § 50.34 (safety analysis report);

(iii) For an application for a license for any other facility, or an amendment to a license for such facility: 19 copies of both that portion of the application containing the information required by § 50.33 (general information) and that portion of the application containing the information required by § 50.34 (safety analysis report).

(iv) ***

(2) With respect to an application for a license described in subdivision (1)(i) of this paragraph, the applicant shall update 10 copies of the application and, upon notification of the appointment of an atomic safety and licensing board to conduct the public hearing required by the Atomic Energy Act for the issuance of a construction permit, serve such updated copies of the application, eliminating all superseded information, together with an index of the updated application, as follows:

(i) Each atomic safety and licensing board member and alternate—one copy.

(ii) Chairman of the Atomic Safety and Licensing Board Panel—one copy.

(iii) Office of the Secretary—one copy.

(iv) Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate—two copies.

39 FR 17972

Any subsequent amendments to the application shall be served in the same manner, and three signed originals and the specified number of copies of such amendments shall be filed with the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, as provided in paragraph (c)(1)(i) of this section.

34 FR 6036

At the time of filing of such an application, one copy shall be made available in an appropriate office near the site of the proposed facility for inspection by the public and updated as amendments to the application prior to the public hearing may be made. This updated copy shall be produced at the public hearing for the use of any other parties to the proceeding. The applicant 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. The applicant shall also update and serve copies of the application and make available a copy of such updated application in an appropriate office near the site of the facility for inspection by the public at such time as the Commission may issue a notice of public hearing concerning the issuance of an operating license.

37 FR 15127

(3) The copies required by subparagraphs (1) and (2) of this paragraph (c) need not be filed until the application has been assigned a docket number or docketed, pursuant to § 2.101(a) of this chapter. Ten (10) copies shall be filed to enable a determination as to whether the application is sufficiently complete to permit the assignment of a docket number or docketing, as appropriate.

35 FR 19655

(d) 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 licenge 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.

33 FR 10923

** (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

***Deleted 39 FR 40249.

**Amended 35 FR 19655.

operating license, except as provided in § 170.21 of this chapter.

37 FR 5745

(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 pur-

§ 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.

21 FR 355

33 FR 9704

(f) Information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the permit or license is sought. If the application is for a construction permit, such information shall show that the applicant possesses the funds necessary to cover estimated construction costs and related fuel cycle costs or that the applicant has reasonable assurance of obtaining the necessary funds, or a combination of the two. If the application is for an operating license, such information shall show that the applicant possesses the funds necessary to cover estimated operating costs or that the applicant has reasonable assurance of obtaining the necessary funds, or a combination of the two. With respect to any production or utilization facility of a type described in § 50.21(b) or § 50.22, or a testing facility, the following specific requirements shall apply:

33 FR 9704

If the application is for an operating license, such information shall show that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover the estimated costs of operation for the period of the license or for 5 years, whichever is greater, plus the estimated costs of permanently shutting the facility down and maintaining it in a safe condition. Without limitation on the generality of the foregoing requirements, each application for a construction permit or an operating license submitted by an entity organized for the primary purpose of constructing or operating a facility shall include information showing the legal and financial relationships it has or proposes to have with its stockholders or owners, and their financial ability to meet any contractual obligation to such entity which they have incurred or propose to incur, and any other information necessary to enable the Commission to determine the applicant's financial qualifications.

*

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.

35 FR 19655

(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.

(k)†

39 FR 34394

§ 50.33a Information required for antitrust review.[1]

(a) 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. This information 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.

39 FR 34394

(b) 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) Any person who applies for such a construction permit prior to July 28, 1975 shall submit the document titled "Information Requested by the Attorney General for Antitrust Review" as soon as possible.

§ 50.34 Contents of applications: Technical information.

34 FR 6769

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[1] 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 subparagraphs (2)–(8) of this paragraph, as well as the information required by this subparagraph, in support of the application for a construction permit.

33 FR 18610

(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.[2] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the

***Amended 39 FR 26279.

[1] The reporting requirements contained in §§ 50.33a, 50.55b, 50.80, and Appendix L of Part 50 have been approved by GAO under B-183225 (R0071). This clearance expires 8-31-77.

*Deleted 34 FR 6036.

†Paragraph (k) deleted 38 FR 3955.

[1] 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.

[2] General design criteria for chemical processing facilities are being developed.

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 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; an 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.

(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 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 effluen and radiation exposures within the limi set forth in Part 20 of this chapter.

(4) A final analysis and evaluation the design and performance of stru tures, systems, and components with t objective stated in paragraph (a)(4) this section and taking into account ar pertinent information developed sin the submittal of the preliminary safe analysis report.

Analysis and evaluation o ECCS cooling performance followin postulated loss-of-coolant accident shall be performed in accordance witi the requirements of § 50.46 for facilitie for which a license to operate may b issued after December 28, 1974.

(5) A description and evaluation c the results of the applicant's program including research and development, any, to demonstrate that any safet questions identified at the constructio permit stage have been resolved.

(6) The following information con cerning facility operation:

(i) The applicant's organizationa structure, allocations or responsibilitie and authorities, and personnel qualifica tions requirements.

(ii) Managerial and administrativ controls to be used to assure safe opera tion. Appendix B, "Quality Assuranc Criteria for Nuclear Power Plants an Fuel Reprocessing Plants" sets forth th requirements for such controls for nu clear power plants and fuel reprocessin plants. The information on the control to be used for a nuclear power plant or fuel reprocessing plant shall include discussion of how the applicable require ments of Appendix B will be satisfie

(iii) Plans for preoperational testin and initial operations. The Commissio has developed documents entitled "Guid For The Planning Of Preoperation Testing Programs" and "Guide For Th Planning Of Initial Startup Programs" to help applicants establish adequat plans pursuant to this subsection.

(iv) Plans for conduct of norma operations, including maintenance, su veillance, and periodic testing of struc tures, systems, and components.

(v) Plans for coping with emergen cies, which shall include the items spec ified in Appendix E.

[3] The Guides are available for inspectio at the Commission's Public Document Room 1717 H Street NW., and copies may be ob tained by addressing a request to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

33 FR 18610

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.

34 FR 6036

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

38 FR 22221

(8) A description and plans for implementation of an operator requalification program. The operator requalification program shall, as a minimum, meet the requirements for those programs contained in Appendix A of Part 55 of this chapter.

38 FR 30537

(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 of this chapter, if applicable, at the proposed facility.[3] Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable.

40 FR 58847

§ 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 provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirement 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.

[3] Regulatory Guide 1.17 dated June 1973 describes physical security criteria generally acceptable for the protection of nuclear power reactors against acts of industrial sabotage.

35 FR 18385

(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; and (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.

35 FR 5317

§ 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.

[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.

35 FR 8644

35 FR 5317

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.

(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.

27 FR 12915

(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, sub-

mitted 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 which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall 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 reoccurrence. Operation shall not be resumed until authorized by the Commission.

(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 which must not be exceeded in order to protect the integrity of the physical system which is designed to guard against the uncontrolled release of radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action shall be taken as stated in the technical specification or the affected part of the process, or the entire process if required, shall be shut down, unless such 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 reoccurrence. If a portion of the process or the entire process has been shut down, operation shall not be resumed until authorized by the Commission.

(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 shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. He 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 reoccurrence.

(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 shall 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 reoccurrence.

(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 specification 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 specification until the condition can be met. In the case of either a nuclear reactor or a 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 action taken to preclude reoccurrence.

(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 subparagraphs (1), (2), and (3) of this paragraph (c).

(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.

(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.

(2) The submission of a report to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 6 months of operation, and such other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such 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.*

*Amended 40 FR 58847.

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.*

40 FR 19439

§ 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 per-

21 FR 355

*Amended 40 FR 58847.

mitting 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.

§ 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 except a license authorizing export only pursuant to an agreement for cooperation.

§ 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.

21 FR 355

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) The applicant is technically and financially qualified to engage in the proposed activities in accordance with the regulations in this chapter.

(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.

36 FR 12731

(d) Any applicable requirements of Part 51* have been satisfied.

§ 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:

(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.

35 FR 19655

(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).

§ 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:

(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 19655

§ 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.

(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.

(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 fa-

*Amended 39 FR 26279.

[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.

cility for the primary purpose of producing electric energy for disposition for ultimate public consumption.

§50.44 *

§ 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.

§ 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.

(a)(1) Except as provided in paragraph (a)(2) and (3) of this section, each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding shall be provided with an emergency core cooling system (ECCS) which shall 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 shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models. Conformance with the criteria set forth in paragraph (b) of this section with ECCS cooling performance calculated in accordance with an acceptable evaluation model, may require that restrictions be imposed on reactor operation.

(2) With respect to reactors for which operating licenses have previously been issued and for which operating licenses may issueon or before December 28, 1974:

39 FR 1001

39 FR 2712

(i) The time within which actions required or permitted under this subparagraph (2) must occur shall begin to run on February 4, 1974.

40 FR 8774

(ii) Within six months following the date specified in paragraph (a)(2)(i) of this section an evaluation in accordance with paragraph (a)(1) of this section shall have been submitted to the Director of Regulation of the Atomic Energy Commission. The evaluation shall have been accompanied by such proposed changes in technical specifications or license amendments as may be necessary to bring reactor operation in conformity with paragraph (a)(1) of this section.

(iii) Any licensee may have requested an extension of the six-month period referred to in paragraph (a)(2)(ii) of this section for good cause. Any such request shall have been submitted not less than 45 days prior to expiration of the six-month period, and shall have been accompanied by affidavits showing precisely why the evaluation is not complete and the minimum time believed necessary to complete it. The Director of Regulation of the Atomic Energy Commission shall have caused notice of such a request to be published promptly in the FEDERAL REGISTER; such notice shall have provided for the submission of comments by interested persons within a time period established by the Director of Regulation. If, upon reviewing the foregoing submissions, the Director of Regulation concluded that good cause had been shown for an extension, he may have extended the six-month period for the shortest additional time which in his judgment will be necessary to enable the licensee to furnish the submissions required by paragraph (a)(2)(ii) of this section. Requests for extensions of the six-month period submitted under this subparagraph will have been ruled upon by the Director of Regulation prior to expiration of that period.

40 FR 8774

(iv) Upon submission of the evaluation required by paragraph (a)(2)(ii) of this section (or under paragraph (a)(2)(iii), if the six-month period is extended) the facility shall continue or commence operation only within the limits of both the proposed technical specifications or license amendments submitted in accordance with this paragraph (a)(2) and all technical specifications or license conditions previously imposed by the Atomic Energy Commission, including the requirements of the Interim Policy Statement (June 29, 1971, 36 FR 12248) as amended December 18, 1971, 36 FR 24082).

(v) Further restrictions on reactor operation will be imposed if it is found that the evaluations submitted under paragraphs (a)(2)(ii) and (iii) of this section are not consistent with paragraph (a)(1) of this section and as a result such restrictions are required to protect the public health and safety.

(vi) Exemptions from the operating requirements of paragraph (a)(2)(iv) of this section may be granted for good cause. Requests for such exemption shall be submitted not less than 45 days prior to the date upon which the plant would otherwise be required to operate in accordance with the procedures of said paragraph (a)(2)(iv) of this section. Any such request shall be filed with the Secretary of the Commission, who shall cause notice of its receipt to be published promptly in the FEDERAL REGISTER; such notice shall provide for the submission of comments by interested persons within 14 days following FEDERAL REGISTER publication. The Director of Nuclear Reactor Regulation shall submit his views as to any requested exemption within five days following expiration of the comment period.

39 FR 1001

(vii) Any request for an exemption submitted under subparagraph (vi) of this subparagraph (2) must show, with appropriate affidavits and technical submissions, that it would be in the public interest to allow the licensee a specified additional period of time within which to alter the operation of the facility in the manner required by subparagraph (iv) of this subparagraph (2). The request shall also include a discussion of the alternatives available for establishing compliance with the rule.

39 FR 1001

40 FR 8774

(3) Construction permits may have been issued after December 28, 1973 but before December 28, 1974 subject to any applicable conditions or restrictions imposed pursuant to other regulations in this chapter and the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 FR 12248) as amended (December 18, 1971, 36 FR 24082): Provided, however, that no operating license shall be issued for facilities constructed in accordance with construction permits issued pursuant to this paragraph, unless the Commission determines, among other things that the proposed facility meets the requirements of paragraph (a)(1) of this section.

(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.

39 FR 1001

(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 accept-

*Deleted 39 FR 40249.

ably 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.

38 FR 1001

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.

21 FR 355

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.

§ 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 except insofar as the export of production or utilization facilities is authorized.

§ 50.54 Conditions of licenses.

Whether stated therein or not, the following shall be deemed conditions in every license issued:

*(a) [Reserved]

(b) No right to the special nuclear material shall be conferred by the license except as may be defined by the license.

(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.

(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.

21 FR 355

(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 will at any time before expiration of the license, upon request of the Commission submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license should be modified, suspended or revoked.

(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.

(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.

28 FR 3196

(i) Except as provided in § 55.9 of this chapter, the licensee shall 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.

38 FR 22221

(i-1) Within three (3) months after issuance of an operating license, the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A of Part 55 of this Chapter. Notwithstanding the provisions of § 50.59 the licensee shall not,** except as specifically authorized by the Commission, make a change in an approved operator requalification program by which the scope, time allotted for the program or frequency in conducting different parts of the program is decreased.

40 FR 8774

Holders of operating licenses in effect on September 17, 1973 shall implement an operator requalification program which, as a minimum, meets the requirements of Appendix A of Part 55 of this chapter which was submitted for approval by the Atomic Energy Commission.

(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.

(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.

28 FR 3196

(m) 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.

***(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.

38 FR 4385

(o) Primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J.

*Deleted 32 FR 2562.

** Amended 38 FR 26354.

***Revised 31 FR 3668.

38 FR 30537

(p) The licensee shall make no change which would decrease the effectiveness of a security plan prepared pursuant to § 50.34(c) or paragraph (q) of this section without the prior approval of the Commission. A licensee desiring to make such a change shall submit an application for a change in the technical specifications incorporated in his license or for an amendment to his license pursuant to § 50.90, as appropriate. The licensee shall maintain records of changes to the plan made without prior Commission approval, and shall furnish to the Commission a report containing a description of each change within two months after the change is made.

(q) Each licensee who is authorized to operate a production or utilization facility and who has not submitted a physical security plan, as described in § 50.34(c) by November 6, 1973, shall have submitted such a plan to the Atomic Energy Commission for approval by January 7, 1974.

21 FR 355

§ 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. If the construction or modification is completed before the earliest date specified, the holder of the permit shall promptly notify the Commission for the purpose of accelerating final inspection. **

(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.

36 FR 11423

(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.

35 FR 19655

(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 § 50.30(d).

37 FR 6459

(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; or

(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 promptly notify the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office of each reportable deficiency.

(3) The holder of a construction permit shall also submit a written report on a reportable deficiency within 30 days to the Division of Inspection and Enforcement with a copy to the Inspection and Enforcement Regional Office.

The report shall 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, together with a statement as to 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.

36 FR 11423

§ 50.55a Codes and standards.

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.

38 FR 19012

(2) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission or the Atomic Energy Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing, or inspection of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

36 FR 11423

*(ii) Proposed alternatives to the described requirements or portions thereof will provide an acceptable level of quality and safety. For example, the use of inspection or survey systems other than those required by the specified ASME Codes and Addenda may be authorized under this subparagraph provided that an acceptable level of quality and safety in design, fabrication, installation, and testing is achieved.

39 FR 5773

(b) As used in this section, references [1] to editions of Criteria, Codes, and Standards include only those editions through 1971; references [1] to Addenda include only those Addenda through the Summer 1973 Addenda.

37 FR 17021

(c) Pressure vessels:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, pressure vessels which are part of the reactor coolant pressure boundary [2] shall meet the requirements for Class A vessels set forth in section III of the ASME Boiler and Pressure Vessel Code, applicable Code Cases, and Addenda [3] in effect [4] on the date of order [5] of the vessel. The pressure vessels may meet the requirements set forth in editions of this Code, applicable Code Cases, and Addenda which have become effective after the date of vessel order.

(2) For construction permits issued on or after January 1, 1971, pressure vessels which are part of the reactor coolant pressure boundary [2] shall meet the requirements for Class A or Class 1 vessels set forth in editions of section III of the ASME Boiler and Presure Vessel Code

**Deleted 32 FR 4055.

*Redesignated 37 FR 17021.

See next page for footnotes 1 to 5.

and Addenda[6] in effect[4] on the date of order[5] of the pressure vessel: *Provided, however,* That if the pressure vessel is ordered more than 18 months prior to the date of issuance of the construction permit, compliance with the requirements for Class A or Class 1 vessels set forth in editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 18 months prior to the date of issuance of the construction permit is required. The pressure vessels may meet the requirements set forth in editions of this Code and Addenda which have become effective after the date of vessel order or after 18 months prior to the date of issuance of the construction permit.

(d) Piping:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, piping which is part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in:

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases[3] or the U.S.A. Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases[3] or the Class I section of the U.S.A. Standard Code for Pressure Piping (USAS B31.7)[6] in effect[4] on the date of order[5] of the piping and

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards of Class I piping of the U.S.A. Standard Code for Pressure Piping (USAS B31.7) may be applied.

The piping may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and USAS B31.7, Addenda, and Code Cases which became effective after the date of order of the piping.

(2) For construction permits issued on or after January 1, 1971, piping which is part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I piping set forth in editions of (i) the USA Standard Code for Pressure Piping (USAS B31.7) and Addenda[6] in effect[4] on the date of order[5] of the piping and the requirements applicable to piping of articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect on the date of order of the piping, or (ii) the requirements applicable to Class 1 piping of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of the order of the piping; *Provided, however,* That if the piping is ordered more than 6 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I+ or Class 1 piping set forth in editions of USAS B31.7 or section III of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect 6 months prior to the date of issuance of the construction permit is required. The piping may meet the requirements set forth in editions of these Codes and Addenda[6] which have become effective after the date of piping order or after 6 months prior to the date of issuance of the construction permit.

(e) Pumps:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, pumps which are part of the reactor coolant pressure boundary[2] shall meet—

(i) The requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases[6] in effect[4] on the date of order[5] of the pumps, or

(ii) The nondestructive examination and acceptance standards set forth in ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps may be applied.

The pumps may meet the requirements set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which became effective after the date of order of the pumps.

(2) For construction permits issued on or after January 1, 1971, pumps which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I pumps set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] in effect[4] on the date of order[5] of the pumps and the requirements applicable to pumps set forth in articles 1 and 8 of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps, or (ii) the requirements applicable to Class 1 pumps of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps: *Provided, however,* That if the pumps are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] and the requirements applicable to pumps set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda, or for Class I+ pumps of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The pumps may meet the requirement set forth in editions of these Codes or Addenda which have become effective after the date of pump order or after 12 months prior to the date of issuance of the construction permit.

(f) Valves:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases, or the USA Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases, in effect[4] on the date of order[5] of the valves or the Class I section of the Draft ASME Code for Pumps and Valves for Nuclear Power,[3] Addenda, and Code Cases in effect on the date of order of the valves; or

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N2, N7, N9, and N10, except that the acceptance standards for Class I valves set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the valves may be applied.

The valves may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases, which became effective after the date of order of the valves.

(2) For construction permits issued on or after January 1, 1971, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I valves set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] in effect[4] on the date of order[5] of the valves and the requirements applicable to valves set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda[6]

[1] These incorporation by reference provisions were approved by the Director of the Federal Register on March 17, 1972 and May 4, 1973.†

[2] Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary defined in § 50.2(v) need not meet these requirements, provided:

(a) 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 only, or

(b) The component is or can be isolated from the reactor coolant system by two valves (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] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. NW., Washington, DC.

[4] ASME and U.S. of America Standard Code Addenda are considered "in effect" 6 months after their date of issuance.

[5] The Code issue applicable to a component is governed by the order or contract date for the component, not the contract date for the nuclear energy system.

[6] The use of specific Code Cases may be authorized by the Commission upon request pursuant to § 50.55a(a)(2)(ii).*

*Amended 37 FR 18073.
†Amended 38 FR 19907.
+Amended 39 FR 5773.

in effect on the date of order of the valves, or (ii) the requirements applicable to Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valve; *Provided, however,* That if the valves are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I valves set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[6] and the requirements applicable to valves set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda or for Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The valves may meet the requirements set forth in editions of these Codes or Addenda which have become effective after the date of valve order or after 12 months prior to the date of issuance of the construction permit.

37 FR 1702

(g) Inservice inspection requirements: For construction permits issued on or after January 1, 1971, systems and components shall meet the requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda[3,6] in effect 6 months prior to the date of issuance of the construction permit. Systems and components may meet the requirements set forth in editions of this Code and Addenda which have become effective after 6 months prior to the date of issuance of the construction permit.

(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[7] 12 months prior to the date of issuance of the construction permit. Protection systems may meet the requirements set forth in editions or revisions of IEEE 279 which have become effective after 12 months prior to the date of issuance of the construction permit.

38 FR 19012

(i) Fracture toughness requirements: Pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

(j) Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970, may meet the requirements of paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) of this section instead of paragraphs (c)(2), (d)(2), (e)(2), and (f)(2) of this section, respectively.

38 FR 19012

See preceding page for footnotes 3 to 6.

[7] For purposes of this regulation, the proposed IEEE 279 became "in effect" on August 30, 1968, and the revised issue IEEE 279–1971 became "in effect" on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

39 FR 5773

+Amended 39 FR 5773.

§ 50.55b Conditions of construction permits and operating licenses pertaining to antitrust matters.

The Commission may incorporate, in construction permits for production or utilization facilities of the type described in § 50.22 for which applications were on file on December 19, 1970, and in operating licenses for production or utilization facilities of a type described in §§ 50.22 and 50.21(b)(1), 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 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, a condition to the effect that the license shall be subject to an antitrust review by the Attorney General pursuant to section 105c of the Atomic Energy Act of 1954, as amended; that the licensee shall furnish to the Commission the information requested by the Attorney General, as described in Appendix L to this part; that the Commission may hold a hearing on antitrust matters on the recommendation of the Attorney General or at the request of any person whose interest may be affected by the proceeding; that on the basis of its findings made after such hearing, the Commission will continue, rescind, or amend the license to include such conditions as the Commission deems appropriate; and that the licensee shall comply with any order or license condition made by the Commission pursuant to section 105c of the Atomic Energy Act of 1954, as amended, with respect to the licensed activities.

38 FR 3955

§ 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.

§ 50.57 Issuance of operating licenses.[1]

(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; and

(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.

35 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 find-
ings on the matters specified in paragraph (a) of this section and to issue a license for the requested operation.

37 FR 1527

(d) *

[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.

35 FR 6644

* Deleted 40 FR 8774.

§ 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.

39 FR 10554

(b) 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 which is a testing facility. 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. If the Commission finds that no significant hazards consideration is presented by an application for an amendment to a construction permit or operating license, it may dispense with such notice and publication and may issue the amendment.

33 FR 8587

§ 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.

39 FR 10554

(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 of 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) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or 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 shall 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. The licensee shall furnish to the Commisison, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record of the licensing proceeding. In addition to a signed original, 39 copies of each report of changes in a facility of the type described in § 50.21(b) or § 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed.

39 FR 10554

(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.

39 FR 40249

§ 50.60 *

EXPORT LICENSES

§ 50.65 Export of production and utilization facilities.[1] *

(a) *Filing of application.* (1) Each application for a license to export a production or utilization facility, or amendment thereof, should be filed with the Director of Nuclear Materials Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street, NW., Washington, D.C.; at 7920 Norfolk Avenue, Bethesda, Maryland.

39 FR 40249

(2) Each application for a license to export a production or utilization facility, or amendment thereof, should be executed in three signed originals by the applicant or duly authorized officer thereof, under oath or affirmation.

(3) Each filing of an application for a license to export a production or utilization facility, or amendment thereof, should include three copies in addition to the three signed originals.

(b) *Contents of applications: general information.* Each application for a license to export a production or utilization facility shall state:

(1) Name of applicant;

(2) Address of applicant;

(3) Description of business or occupation of applicant;

(4) (i) If applicant is an individual, state citizenship;

(ii) If applicant is a partnership, state name, citizenship, and address of each partner and the principal location where the partnership does business;

(iii) If applicant is a corporation or an unincorporated association, state (A) the State where it is incorporated or organized and the principal location where it does business; and (B) whether it is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government, and if so, give details;

(iv) 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;

(5) Name and address of persons or organizations who are arranging for export, if different from the applicant;

(6) Name and address of ultimate consignee;

(7) Name and address of intermediate consignee(s) or parties to export;

(8) Date of proposed first shipment;

(9) Date of proposed completion of shipment;

(10) Proposed criticality date or date of start of operation;

(11) Proposed expiration date of export license;

(12) Total value of all items to be exported from the United States under the license being applied for.

(c) *Contents of application; additional information for utilization facilities.* Each application for a license to export a utilization facility shall contain the following:

(1) General information:

(i) Type of facility,

(ii) Design power level, if a nuclear reactor, in terms of thermal and (where appropriate) electrical watts or kilowatts,

(iii) Name by which the facility is or will be known,

(iv) Location where the facility is to be installed or built.

(2) A list of structures, systems, or components to be exported which are associated with the construction, maintenance, and operation of the utilization facility proposed for export and which fall within the categories described in paragraphs (c)(2)(i) through (vi) of this section. Such list need only identify the items by appropriate category titles in lieu of a specific item-by-item list.

(i) *Reactor coolant pressure boundary.* Those structures, systems, and components of a nuclear reactor located within or forming a part of the reactor coolant pressure boundary as defined in § 50.2(v).

[1] For other provisions of Part 50 relating to requirements for export licenses, ineligibility of certain applicants for export licenses and jurisdictional limitations, see § 50.10, 50.21, 50.22, 50.38, and 50.53.

* The requirements contained in § 50.65 have been approved by GAO under B-180225 (R0062). The approval expires 5/31/77.

* Deleted 40 FR 8774.

(ii) *Instrumentation.* Instrumentation systems for indication, control, and protection of a nuclear reactor, including their associated equipment, which are directly associated with structures, systems, and components located within or forming part of the reactor coolant pressure boundary and which are normally required for routine startup, power operation, or shutdown of the reactor, or for periodic testing. Portions of other instrumentation systems of the facility mounted in a common panel with the covered systems are included in this category.

(iii) *Fuel handling equipment.* Fuel handling equipment used to load new or recycled fuel into a reactor core, to unload fuel from a reactor core, or to transfer fuel within a reactor facility and place it into a facility povided for onsite storage or into fuel shipping equipment.

(iv) *Experimental facilities.* Experimental facilities whose primary purpose is the irradiation or activation of material by radiation from a nuclear reactor, or which are used with a reactor to provide a source of nuclear radiation for tests or experiments.

(v) *Spare or replacement components.* Spare or replacement components or parts for items in paragraphs (c) (2) (i) through (iv) of this section, which are furnished during duration of the export license as part of the initial purchase or under a warranty from the vendor.

(vi) *Equipment or tools.* Special equipment or tools needed to service, maintain, or replace items in paragraphs (c) (2) (i) through (v) of this section. Equipment or tools falling under this category which are not intended by the applicant to remain with the facility being exported but are intended to be reexported, resold, retransferred, disposed of, or returned to the United States shall be identified and their proposed disposal or retransfer described. Complete names and addresses of parties outside of the United States other than the ultimate consignee named in the application, to whom resale, reexport, retransfer, or other such disposal or disposition of equipment and tools exported in connection with the proposed facility export license is intended, shall be included. The end use and uses of such equipment or tools by each consignee shall be described.

(3) An itemized list of other structures, systems, or components to be exported which are associated with the construction, maintenance, and operation of the utilization facility proposed for export but which do not fall within the categories listed in paragraph (c) (2) of this section. Such itemized list should identify the specific items to be exported and should reflect the commodity control list numbers set forth in the regulations of the Office of Export Administration, U.S. Department of Commerce (see 15 CFR Part 399). The Commission will determine, for each listed item, whether its export should be licensed by the Commission.

(d) *Standards for licenses authorizing export only.* A license authorizing the export of production or utilization facilities may be issued by the Commission upon determining that:

(1) The issuance of the license to the applicant for export of the facility involved is within the scope of and consistent with the terms of an agreement for cooperation pursuant to Section 123 of the Atomic Energy Act of 1954, as amended, and the export would not be inconsistent with U.S. obligations under any treaty or other international agreement; and

(2) The application complies with the requirements of the Atomic Energy Act, as amended, and the regulations set forth in this chapter.

(e) *Conditions of licenses.* The following shall be deemed to be conditions in each facility export license:

(1) No authority to export special nuclear material, byproduct material, or source material shall be conferred by the license.

(2) Neither the license, nor any right thereunder, 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 gives its consent in writing.

(3) The license shall be subject to revocation, suspension, modification, or amendment for cause as provided in the Act and Commission regulations, in accordance with procedures provided by the Act and the regulations in this chapter.

(4) 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.

(5) No items shall be exported under the license unless specifically required for the facility licensed for export.

(6) No items exported under the license shall be reexported or otherwise disposed of by the licensee or used by the licensee for any purpose other than that stated in the application, unless the Commission approves in writing of the disposition or use.

INSPECTION, RECORDS, REPORTS

§ 50.70 Inspections.

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.

§ 50.71 Maintenance of records, making of reports.

(a) Each licensee and each holder of a construction permit shall 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 Act, including section 105 of the Act.

(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, upon each issuance of its annual financial report, including the certified financial statements, file a copy thereof with the Commission.

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 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.

‡Amended 38 FR 3955.

§ 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 notowned 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.

35 FR 19655

(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.

26 FR 9546

§ 50.82 Applications for termination of licenses.

(a) Any licensee may apply to the Commission for authority to surrender a license voluntarily and to dismantle the facility and dispose of its component parts. The Commission may require information, including information as to proposed procedures for the disposal of radioactive material, decontamination of the site, and other procedures, to provide reasonable assurance that the dismantling of the facility and disposal of the component parts 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.

32 FR 3090

(b) If the application demonstrates that the dismantling of the facility and disposal of the component parts 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 may issue an order authorizing such dismantling and disposal, and providing for the termination of the license upon completion of such procedures in accordance with any conditions specified in the order.

26 FR 9546

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 shall be filed with the Commission, fully describing the changes desired, and following as far as applicable the form prescribed for original applications.

21 FR 355

§ 50.91 Issuance of amendment.

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 prior to 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 pursuant to § 2.105 of this chapter before acting thereon. The notice will be issued as soon as practicable after the application has been docketed.

39 FR 13258

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)); 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 Administration finds that the production program of the Administration 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 Administration 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.

40 FR 8774

§ 50.103 *Suspension and operation in war or national emergency.* (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.

21 FR 355

(2) Cause the recapture of special nuclear material.

40 FR 8774

(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.

21 FR 355

BACKFITTING

§ 50.109 Backfitting.

(a) The Commission may, in accordance with the procedures specified in this chapter, require the backfitting of a facility if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security. As used in this section, "backfitting" of a production or utilization facility means the addition, elimination or modification of structures, systems or components of the facility after the construction permit has been issued.

(b) Nothing in this section shall be deemed to relieve a holder of a construction permit or a license from compliance with the rules, regulations, or orders of the Commission.

(c) The Commission may at any time require a holder of a construction permit or a license to submit such information concerning the addition or proposed addition, the elimination or proposed elimination, or the modification or proposed modification of structures, systems or components of a facility as it deems appropriate.

35 FR 5317

ENFORCEMENT

§ 50.110 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, or Title II of the Energy Reorganization Act of 1974, or any regulation or order issued thereunder. A court order may be obtained for the payment of a civil penalty imposed pursuant to section 234 of the Act for violation of section 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Act, or section 206 of the Energy Reorganization Act of 1974, or any rule, regulation, or order issued thereunder, or any term, condition, or limitation of any license issued thereunder, or for any violation for which a license may be revoked under section 186 of the Act. Any person who willfully violates any provision of the Act or any regulation or order issued thereunder may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

NOTE: The record keeping and reporting requirements contained in this part have been approved by the Office of Management and Budget in accordance with the Federal Reports Act of 1942.

APPENDICES

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish 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 been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

36 FR 3255

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.

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the envrionmental 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.

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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.

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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.

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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.

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.

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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.

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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.

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.

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.

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Criterion 18—Inspection and testing of electrical 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.

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.

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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 be designed 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 in

component design and principles of operation, shall be used to the extent practical to prevent loss 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.

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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.

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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.

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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 leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

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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.

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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 leak-tight 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 removal 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 energy from metal-water and other chemical reactions that may result from degraded 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 system shall be prevented by physical systems or processes, preferably by use of geometrically safe

36 FR 3255

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.

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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.

35 FR 10498

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[1] 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.

40 FR 3210C

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.

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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-

[1] 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.

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, or 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 powerplant 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 powerplant or fuel reprocessing plant,* such as by tagging valves and switches, to prevent inadvertent operation.

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 the significant condition 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-

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*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.

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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.

APPENDIX C—A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR FACILITY CONSTRUCTION PERMITS AND OPERATING LICENSES

GENERAL INFORMATION

This appendix is intended to apprise applicants for licenses to construct and operate production or utilization facilities of the types described in § 50.21(b) or § 50.22, or a testing facility, of the general kinds of financial data and other related information that will demonstrate the financial qualifications of the applicant to carry out the activities for which the permit or license 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 and operate the proposed facility.

Since separate findings of financial qualifications will be made by the Commission at the construction permit stage of the licensing process and at the operating license stage, the nature of the information to be included in the application at each of these stages is discussed separately.

It is important to observe also that both § 50.33(f) and this appendix distinguished 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 or license 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.

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In determining an applicant's financial qualifications, 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, at the operating license stage, and during operation of the facility, 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 the sharing of the power from the facility by two or more electrical generating companies.

Applicants, permit holders, and licensees 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 ---------- $_____

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 Power Commission or an explanation given as to any departures 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.

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.

3. *Applicant's financial statements.* The application should also include the applicant's latest published annual financial report, together with such current interim financial statements as are pertinent. If such 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.

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B. *Applications for operating licenses.* An application for a facility operating license will usually be filed near the time of completion of construction of the facility. Section 50.33(f) requires that all applications for operating licenses show that the applicant possesses the funds necessary to cover estimated operating costs, or has reasonable assurance of obtaining the necessary funds, or a combination of the two. In addition, each application for a license for a facility other than a medical or research reactor is required to show that the applicant possesses or has reasonable assurance of obtaining the funds necessary to pay the estimated costs of operation for the period of the license or for 5 years, whichever is greater, plus the estimated costs of permanently shutting down the facility and maintaining it in a safe condition. For purposes of the latter requirement, it will ordinarily be sufficient to show at the time of filing of the application, availability of resources sufficient to cover estimated operating costs for each of the first 5 years of operation plus the estimated costs of permanent shutdown and maintenance of the facility in safe condition. It is also expected that, in most cases, the applicant's annual financial statements contained in its published annual reports will enable the Commission to evaluate the applicant's financial capability to satisfy this requirement.

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 de-

[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.

scribed 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.

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.

B. *Applications for operating licenses*—1. *Current financial statements.* In its application for a license to operate the facility, the applicant should include a statement of its current financial condition.

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2. *Estimates of operating income and expense.* In order to enable the Commission to evaluate the applicant's financial qualifications to operate the completed facility and, if necessary, to shut it down, as required by § 50.33(f), the application for a license to operate a facility other than a medical or research reactor should include a statement of the applicant's estimate of annual income and expense for the first 5 years of operation. The statement should list operating revenues and expense in sufficient detail to permit an assessment of the reasonableness of the estimates. In addition, the applicant should include its estimate of costs to permanently shut down the facility and maintain it in safe condition if that should become necessary.

III. ANNUAL FINANCIAL STATEMENTS

Each licensee and 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.

IV. ADDITIONAL INFORMATION

The Commission may, from time to time, request the applicant or licensee, 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 or a licensee's financial qualifications to continue the conduct of the activities authorized by the license and to shut down the facility and maintain it in safe condition.

APPENDIX D*

* Deleted 39 FR 26279.

APPENDIX E—EMERGENCY PLANS FOR PRODUCTION AND UTILIZATION FACILITIES

I. Introduction

Each applicant for a construction permit is required by § 50.34(a) to include in its 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 its final safety analysis report plans for coping with emergencies.

This appendix establishes minimum requirements for emergency plans. These plans shall be described in the preliminary safety analysis report and submitted as a part of the final safety analysis report. Procedures used in the detailed implementation of emergency plans need not be described in the preliminary or final safety analysis report.

II. The Preliminary Safety Analysis Report

The Preliminary Safety Analysis Report shall contain sufficient information to assure the compatibility of proposed emergency plans with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, and land use.

As a minimum, the following items shall be described:

A. The organization for coping with emergencies, and the means for notification, in the event of an emergency, of persons assigned to the emergency organization;

B. Contacts and arrangements made or to be made with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies;

C. Measures to be taken in the event of an accident within and outside the site boundary to protect health and safety and prevent damage to property and the expected response in the event of an emergency, of offsite agencies;

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D. Features of the facility to be provided for onsite emergency first aid and decontamination, and for emergency transportation of individuals to offsite treatment facilities;

E. Provisions to be made for emergency treatment of individuals at offsite facilities;

F. The training program for employees and for other persons, not employees of the licensee, whose services may be required in coping with an emergency;

G. Features of the facility to be provided to assure the capability for plant evacuation and the capability for facility reentry in order to mitigate the consequences of an accident or, if appropriate, to continue operation.

III. The Final Safety Analysis Report

The Final Safety Analysis Report shall contain plans for coping with emergencies. The details of these plans and the details of their implementation need not be included, but the plans submitted must include a description of the elements set out in section IV to an extent sufficient to demonstrate that the plans provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect public health and safety and prevent damage to property.

IV. Content of Emergency Plans

The emergency plans shall contain, but not necessarily be limited to, the following elements:

A. The organization for coping with radiation emergencies, in which specific authorities, responsibilities, and duties are defined and assigned, and the means of notification, in the event of an emergency, of: (1) Persons assigned to the licensee's emergency organization, and (2) appropriate State, and Federal agencies with responsibilities for coping with emergencies;

B. Written identification, by position or function, of other employees of the licensee with special qualifications for coping with emergency conditions which may arise. Other persons with special qualifications who are not employees of the licensee and who may be called upon for assistance shall also be identified. The special qualifications of these employees and persons shall be described;

C. Means for determining the magnitude of the release of radioactive materials, including criteria for determining the need for notification and participation of local and State agencies and the Nuclear Regulatory Commission and other Federal agencies, and criteria for determining when protective measures should be considered within and outside the site boundary to protect health and safety and prevent damage to property;

D. Procedures for notifying, and agreements reached with, local, State, and Federal officials and agencies for the early warning of the public and for public evacuation or other protective measures should such warning, evacuation, or other protective measures become necessary or desirable, including identification of the principal officials, by title and agencies;

E. Provisions for maintaining up to date: 1. The organization for coping with emergencies, 2. the procedures for use in emergencies, and 3. the lists of persons with special qualifications for coping with emergency conditions;

F. Emergency first aid and personnel decontamination facilities, including:

1. Equipment at the site for personnel monitoring;

2. Facilities and supplies at the site for decontamination of personnel;

3. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

4. Arrangements for the services of a physician and other medical personnel qualified to handle radiation emergencies; and

5. Arrangements for transportation of injured or contaminated individuals to treatment facilities outside the site boundary;

G. Arrangements for treatment of individuals at treatment facilities outside the site boundary;

H. Provisions for training of employees of the licensee who are assigned specific authority and responsibility in the event of an emergency and of other persons whose assistance may be needed in the event of a radiation emergency;

I. Provisions for testing, by periodic drills, of radiation emergency plans to assure that employees of the licensee are familiar with their specific duties, and provisions for participation in the drills by other persons whose assistance may be needed in the event of a radiation emergency;

J. Criteria to be used to determine when, following an accident, reentry of the facility is appropriate or when operation should be continued.

The Commission has developed a document entitled "Guide To the Preparation of Emergency Plans for Production and Utilization Facilities"[1] to help applicants establish adequate plans required pursuant to § 50.34 and this Appendix, for coping with emergencies.

[1] The Guide is available for inspection at the Commission's Public Document Room, 1717 H Street NW. and copies may be obtained by addressing a Request to the Director of Nuclear Reactor Regulation or Director of Nuclear Materials Safety and Safeguards, as appropriate, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

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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. NRC 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.

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.

35 FR 17530

6. With respect to fuel reprocessing plants already licensed, the licenses 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.

36 FR 5411

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APPENDIX G—FRACTURE TOUGHNESS REQUIREMENTS

I. INTRODUCTION AND SCOPE

This appendix specifies minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water cooled 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 requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi.

B. Welds and weld heat-affected zones in the materials specified in section I.A.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi.

Adequacy of the fracture toughness of other ferritic materials shall be demonstrated to the Commission on an individual case basis.

II. DEFINITIONS

A. "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" (unless another section is specified), 1971 Edition, and addenda through the Winter, 1972 Addenda.[1]

B. "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 structure.

C. "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, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems."

D. "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 pursuant to § 50.55a.

E. "Lowest service temperature" means the lowest service temperature as defined by paragraph NB-2332 of the ASME Code.

F. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in paragraph NB-2331 of the ASME Code.

G. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Appendix H) by adding to RT_{NDT} the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50 ft lb level or measured at the 35 mil lateral expansion level, whichever temperature shift is greater.

H. "Beltline region of reactor vessel" means the shell material (including welds and weld heat-affected zones) that directly surrounds the effective height of the fuel element assemblies and any additional height of shell material for which the predicted adjustment of reference temperature at end of service life of the reactor vessel exceeds 50° F.

I. "Material surveillance program" means the provisions for the placement of reactor vessel beltline material specimens in the reactor vessel, and the program of periodic withdrawal and testing of such specimens to monitor, over the service life of the vessel, changes in the fracture toughness properties of the beltline as a result of exposure to neutron irradiation and the thermal environment.

J. "Integrated surveillance programs" means the combination of individual material surveillance programs as applied to one or more reactor vessels to yield results which serve to monitor the changes in fracture toughness properties for a group of vessels.

III. FRACTURE TOUGHNESS TESTS

A. To demonstrate compliance with the minimum fracture toughness requirements of sections IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code, section NB-2300, "Fracture toughness requirements for materials." Both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials shall be tested by means of the dropweight test specified by paragraph NB-2321.1 of the ASME Code. Provision shall be made for supplemental tests in crucial situations such as that described in Section V.C.

B. Charpy V-notch impact tests and dropweight tests shall be conducted in accordance with the following requirements:

1. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens shall be representative of the actual materials of the finished component as required by the applicable rules of the construction code under which the component is built pursuant to § 50.55a, except that ferritic materials intended for the reactor vessel beltline region shall comply with the additional requirements of section III.C. of this appendix.

3. Calibration of temperature instruments and Charpy V-notch impact test machines used in impact testing shall comply with the requirements of paragraph NB-2360 of the ASME Code.

4. Individuals performing fracture toughness tests shall be qualified by training and experience and shall have demonstrated competency to perform the tests in accord with written procedures of the component manufacturer.

5. Fracture toughness test results shall be recorded and shall include a certification by the licensee or person performing the tests for the licensee that:

a. The tests have been performed in compliance with the requirements of this appendix,

b. The test data are correctly reported and identified with the material intended for a pressure-retaining component,

c. The tests have been conducted using machines and instrumentation with available records of periodic calibration, and

d. Records of the qualifications of the individuals performing the tests are available upon request.

C. In addition to the test requirements of section III.A. of this appendix, tests on materials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements:

1. Charpy V-notch (C_v) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_v test curves (including the upper-shelf levels) in terms of both fracture energy and lateral expansion of specimens. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in the vessel shell course(s) following completion of the production longitudinal weld joint, and subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process, in accordance with paragraph NB-2211 of the ASME Code. Where seamless shell forgings are used, or where the same welding process is used for longitudinal and circumferential welds in plates, the test specimens may be taken from a separate weldment provided that such a weldment is prepared using excess material from the shell forging(s) or plates, as applicable, the same heat of filler material, and the same production welding conditions as those used in joining the corresponding shell materials.

IV. FRACTURE TOUGHNESS REQUIREMENTS

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials shall meet the following requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. The materials shall meet the acceptance standards of paragraph NB-2330 of the ASME Code, and the requirements of sections IV.A.2, 3 and 4 and IV.B. of this appendix.

2. For vessels, exclusive of bolting or other fasteners:

a. Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code Appendix G, "Protection Against Non-Ductile Failure". The calculation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture, making appropriate allowance for all uncertainties in the data and analyses.

b. For nozzles, flanges and shell regions near geometric discontinuities, the data and procedures required in addition to those specified in the ASME Code shall provide margins of safety comparable to those required for shells and heads remote from discontinuities.

c. Whenever the core is critical, the metal temperature of the reactor vessel shall be high enough to provide an adequate margin of protection against fracture, taking into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. In no case when the core is critical (other than for the purpose of low-level physics tests) shall the temperature of the reactor vessel be less than the minimum permissible temperature for the inservice system hydrostatic pressure test nor less than 40°F above that temperature required by section IV.A.2.a.

d. If there is no fuel in the reactor during the initial preoperational system leakage and hydrostatic pressure tests, the minimum permissible test temperature shall be determined in accordance with paragraph G2410 of the ASME Code except that the factor of safety applied to each term making up the calculated stress intensity factor may be reduced to 1.0. In no case shall the test temperature be less than $RT_{NDT}+60°F$.

3. Materials for piping (i.e., pipe, tubes and fittings), pumps, and valves (excluding bolting materials) shall meet the requirements of paragraph G3100 of the ASME Code.

4. Materials for bolting and other fasteners with nominal diameters exceeding 1 inch shall meet the minimum requirements of 25 mils lateral expansion and 45 ft lbs in terms of Charpy V-notch tests conducted at the preload temperature or at the lowest service temperature, whichever temperature is lower.

B. Reactor vessels beltline materials shall have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirra-

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. N.W., Washington, D.C.

diated specimens in accordance with paragraphs NB-2322.2(4) and 2322.2(6) of the ASME Code, of 75 ft lbs unless it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper shelf fracture energy are adequate.

C. Reactor vessels for which the predicted value of adjusted reference temperature exceeds 200°F shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. INSERVICE REQUIREMENTS—REACTOR VESSEL BELTLINE MATERIAL

A. The properties of reactor vessel beltline region materials, including welds, shall be monitored by a material surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements" set forth in Appendix H.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of section IV.A.2. are satisfied, using the predicted value of the adjusted reference temperature at the end of the service period to account for the effects of irradiation on the fracture toughness of the beltline materials. The basis for the prediction shall include results from pertinent radiation effects studies in addition to the results of the surveillance program of section V.A.

C. In the event that the requirements of section V.B. cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. An esentially complete volumetric examination of the beltline region of the vessel including 100 percent of any weldments shall be made in accordance with the requirements of Section XI of the ASME Code.

2. Additional evidence of the changes in fracture toughness of the beltline materials resulting from exposure to neutron irradiation shall be obtained from results of supplemental tests, such as measurements of dynamic fracture toughness of archive material that has been subjected to accelerated irradiation.

3. A fracture analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of adequate margins for continued operation.

D. If the procedures of section V.C. do not indicate the existence of an adequate safety margin, the reactor vessel beltline region shall be subjected to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishment of the adjusted reference temperature 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.2., using the values of adjusted reference temperature that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of sections V.C. and V.D. shall be reported to the Commission for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of section V.B.

38 FR 10912

APPENDIX H—REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

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 water cooled power reactors resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

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 constructed of ferritic materials which do not meet the conditions of section II.A. shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, ASTM Designation: E-185-73,[1] except as modified by this appendix.

C. The surveillance program shall meet the following requirements:

1. Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by section III of Appendix G. The specimen types shall comply with the requirements of section III.A. of Appendix G (except that drop weight specimens are not required).

2. Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules, for which the calculated neutron flux will exceed three times the calculated maximum neutron flux at the inside wall of the vessel, may be used in addition to the required number of surveillance capsules specified in section II.C.3.

3. The required number of surveillance capsules and their withdrawal schedules are as follows:

a. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel, making appropriate allowances for all uncertainties in the measurements, that the adjusted reference temperature established in accordance with section IV.B. will not exceed 100°F at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—One-fourth service life
Second capsule—Three-fourths service life
Third capsule—Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the reference temperature greater than originally predicted for similar material as recorded in the applicable technical specification, the remaining withdrawal schedule shall be modified as follows:

REVISED WITHDRAWAL SCHEDULE

Second capsule—One-half service life
Third capsule—Standby

b. For reactor vessels which do not meet the conditions of section II.C.3.a. but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted reference temperature will not exceed 200°F at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second capsule—At approximately one-half of the time interval between first and third capsule withdrawal.
Third capsule—Three-fourths service life.
Fourth capsule—Standby.

c. For reactor vessels which do not meet the conditions of section II.C.3.b., at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second and third capsules—At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.
Fourth capsule—Three-fourths of service life.
Fifth capsule—Standby.

d. Provision shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

e. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

f. If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

g. Proposed withdrawal schedules that differ from those specified in paragraphs e. through f. shall be submitted, with a technical justification therefor, to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

4. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

III. FRACTURE TOUGHNESS TESTS

A. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of section III of Appendix G, "Frac-

38 FR 19012

[1] Effective March 1, 1973. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103, either as a separate or (when available) as part of the 1973 Annual Book of ASTM Standards, Part 30 and also in Part 31. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. NW., Washington, D.C.

ture Toughness Requirements."

B. The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of section V.B. of Appendix G are satisfied.

IV. REPORT OF TEST RESULTS

38 FR 19012

A. Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be provided to the Commission. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the relationship of the measured results to those predicted for the reactor vessel beltline region.

B. The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values of fluence.

C. The operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes made in operational procedures to assure meeting such temperature limitations.

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 is reasonably achievable.*

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 is reasonably achievable.*

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.

40 FR 19439

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.

40 FR 19439

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.

40 FR 40815

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.

40 FR 19439

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 estimation 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

[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.

*Amended 40 FR 58847.

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;

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.

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.*

40 FR 19439

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.

A. If the quantity of radioactive material actualy 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: [2]

1. Make an investigation to identify the causes for such release rates;

2. Define and initiate a program of corrective action; and

3. Report these actions to the Commission within 30 days from the end of the quarter during which the release occurred.

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.

40 FR 19439

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:

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.*

40 FR 40816

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2)

GUIDES ON DESIGN OBJECTIVES FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

A. For radioactive material above background [3] 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 [4] 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:

a. the applicant submits an evaluation of the potential for effects from long-term buildup in the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

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

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

*Amended 40 FR 58847.

[2] Section 50.36a(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 licensee to take such action as the Commission deems appropriate.

[3] "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.

[4] 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.

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.

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 [5] 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 FR 40816

[5] Such in-plant control measures may include treatment of steam generator blow-down 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.

*Amended 40 FR 58847.

APPENDIX J

PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

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 FR 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.

38 FR 4385

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.3.(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.4.(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 maloperating 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.[1] The method chosen for the initial test shall normally be used for the periodic tests.

(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 of not more than La (Ltm/Lam). In the event Ltm/Lam is greater than 0.7, Lt shall be specified as equal to La $(Pt/Pa)^{1/2}$.

(2) *Peak presure 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 value 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 that 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 seal-water 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 inservice inspections.[2]

(b) Permissible periods for testing. The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under the administrative control and in accordance with the safety procedures defined in the license.

2. *Type B tests.* Type B tests except tests for air locks, shall be performed during each

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[1] ANSI N45.4–1972 Leakage Rate Testing of Containment Structures for Nuclear Reactors (dated Mar. 16, 1972). Copies may be obtained from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, IL 60521. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, DC. The incorporation by reference was approved by the Director of the Federal Register on October 30, 1972.

*Amended 38 FR 5997.

[2] Such inservice inspections are required by § 50.55a.

reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years. Air locks shall be tested at 6-month intervals. However, air locks which are opened during such intervals, shall be tested after each opening. 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.

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 shall be the subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of each test. The report shall 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.3, 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.

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* Amended 38 FR 5997.

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 Reaction," 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

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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 nonfuel 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, 1965). 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 or 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. Enging. 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.

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), 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 the Fluid at Low Qualities,: MIT Report Number 9079-26, Cambridge, Massachusetts, September 1963); and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," U.S.N.R.C. 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, "Partial Film Boiling with Water at 2000 psig in a Round Vertical Tube," MSA Research Corp., Technical Report 62 (NP-6976), (1958) is suitable for use between nucelate and film boiling. Use of all these correlations shall be restricted as follows:

(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.

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.

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^{-1}-ft^{-2}°F^{-1} 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^{-1}-ft^{-2}°F^{-1} 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 7x7 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^{-1}-ft^{-2}-°F^{-1} 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 Accident 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 in difference form, the assumptions 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. The description shall be sufficiently detailed and specific to require significant changes in the evaluation model to be specified in amendments of the description. For this purpose, a significant change is a change that would result in a calculated fuel cladding temperature different by more than 20°F from the temperature calculated (as a function of time) for a postulated LOCA using the last previously accepted model.

c. A complete listing of each computer program, in the same form as used in the evaluation model, shall be furnished to the Nuclear Regulatory Commission.

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 compliance with required features of Section I of this Appendix K and provision of a level of safety and margin of conservatism comparable to other acceptable evaluation models, taking into account significant differences in the reactors to which they apply.

39 FR 1001

38 FR 3955

APPENDIX L—INFORMATION REQUESTED BY THE ATTORNEY GENERAL FOR ANTITRUST REVIEW FACILITY LICENSE APPLICATIONS

39 FR 34394

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 and not less than twenty-five (25) additional copies shall be retained by the applicant to be available as needed during the antitrust review.

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 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.

38 FR 3955

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 loan)[1] by contracting parties or pool for the period when the proposed unit will first come on line.

38 FR 3955

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),[2] 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.

39 FR 34394

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.

38 FR 3955

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.

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.

38 FR 3955

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

[1] Indicate whether loads other than peak loads are considered.

[2] List separately and identify certificates of concurrence.

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.

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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 components) 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.

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 3025

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.

2. An application for a manufacturing license pursuant to this Appendix M shall 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 shall be specified in the application. Such application shall 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).

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 Part 51,* 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 detailed statements of environmental considerations prepared by the Commission's regulatory staff will be similarly directed.

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.

(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.

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:

38 FR 3025

(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

*Amended 39 FR 26279.

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.

(g) On the basis of the evaluations and analyses of the environmental effects of the proposed action required by Part 51* 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.

38 FR 30251

8. The prohibition in § 50.10(c) 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 7.

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 operat. 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.

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 Part 51* 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.

38 FR 30251

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 M, 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.

* Amended 39 FR 26279.

APPENDIX N—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OF 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 2976

2. Applications for construction permits submitted pursuant to this Appendix N shall include the information required by §§ 50.33, 50.33a, 50.34(a) and 50.34a(a) and (b). The applicant shall also submit the information required by § 51.20 of this chapter.

For the technical information required by §§ 50.34(a)(1)–(5) and (8) and 50.34a(a) and (b), reference may be made to a single preliminary safety analysis of the design[2] 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 5034a(c). The applicant shall also submit the information required by § 51.21 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.

[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.

[2] 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 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 shall be made in the same manner and in the same number of copies as provided in § 50.30(a), (c)(1) and (3) for license applications.

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 of 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 2976

4. Once the 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 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 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 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 Board, atomic safety and licensing boards, and other presiding officers in any proceeding under Subpart G of Part 2 of this chapter.

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 § 51.5 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.

40 FR 2976

NOTE: The reporting and application requirements contained in Appendix I have been approved by GAO under B-180225 (R0071). The approval expires June 30, 1978.

Title 10—Atomic Energy

CHAPTER 1—ATOMIC ENERGY COMMISSION

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Interim Policy Statement on Implementation, Federal Water Pollution Control Act Amendments of 1972

On October 18, 1972, the Federal Water Pollution Control Act Amendments of 1972, Public Law 92-500, 86 Stat. 816 (hereinafter FWPCA), became law. These amendments completely restructured the Federal Water Pollution Control Act previously in effect and also modified Federal agencies' responsibilities and authorities under the National Environmental Policy Act of 1969 (hereinafter NEPA). Section 511 of the Federal Water Pollution Control Act as restructured provides in pertinent part as follows:

SEC. 511. (a) This Act shall not be construed as (1) limiting the authority or functions of any officer or agency of the United States under any other law or regulation not inconsistent with this Act * * *

* * * * * *

SEC. 511. (c)

(2) Nothing in the National Environmental Policy Act of 1969 (83 Stat. 852) shall be deemed to—

(A) Authorize any Federal agency authorized to license or permit the conduct of any activity which may result in the discharge of a pollutant into the navigable waters to review any effluent limitation or other requirement established pursuant to this Act or the adequacy of any certification under section 401 of this Act; or

(B) Authorize any such agency to impose, as a condition precedent to the issuance of any license or permit, any effluent limitation other than any such limitation established pursuant to this Act.

Set forth below is a Commission interim statement of policy concerning the effect of section 511 of the FWPCA upon the Commission's regulatory responsibility and authority under NEPA in licensing actions covered by 10 CFR Part 50, Appendix D. In developing this interim statement of policy the Commission has viewed the provisions of the FWPCA in conjunction with the mandate of NEPA, as construed in Calvert Cliffs' Coordinating Committee v. AEC, 449 F. 2d 1109 (D.C. Cir. 1971), and embodied in 10 CFR Part 50, Appendix D, i. e., that in regard to major Federal actions having a significant effect on the environment, environmental costs be evaluated and balanced along with benefits and that alternatives be considered that could affect this balancing. The Commission has paid particular attention to the interrelationship between sections 511(a)(1) and 511(c)(2) and the interim statement of policy has as its central premise that AEC's authority and responsibility under NEPA (as implemented by Appendix D to 10 CFR Part 50) remain unaffected except to the extent that there is a conflict with implementing actions taken under the FWPCA. In general, the Commission would continue to exercise its NEPA authority and responsibility in the interim period before various implementing actions are taken under the FWPCA, so that there would be no hiatus in Federal responsibility and authority respecting environmental matters embraced by both NEPA and FWPCA. In addition, the Commission has been mindful of section 101(f) of the FWPCA and has endeavored to avoid to the maximum extent possible needless duplication of regulatory effort.

In summary, the interim statement of policy provides as follows:

(1) If and to the extent that there are applicable limitations or other requirements imposed pursuant to the FWPCA, the Commission will not (with certain exceptions relating to matters of State law) impose different limitations or requirements pursuant to NEPA as a condition to any license or permit. The Commission will itself determine compliance with limitations or requirements promulgated pursuant to FWPCA where no prior compliance determination has been made under FWPCA or where a certain type of interim certification under section 401 of FWPCA has been provided.

(2) The Commission will not consider various alternatives where such action would constitute a review of similar consideration of alternatives under FWPCA and upset a limitation or requirement imposed as a result thereof or where a particular alternative has been required to be adopted pursuant to FWPCA.

(3) In considering the costs and benefits of a proposed action pursuant to NEPA, the Commission will continue to evaluate and give full consideration to environmental impact: *Provided* That, with certain exceptions relating to matters of State law, such evaluation and consideration will be conducted on the basis of discharges or other activities which are at the level of limitations or requirements promulgated or imposed pursuant to FWPCA.

To the extent that there is a conflict between any of the provisions of the interim statement of policy and the provisions of 10 CFR Part 50, Appendix D, the provisions of the statement will govern.

Because these modifications to the Commission's regulations implementing NEPA are necessary to comply with the FWPCA, the Commission has found that good cause exists for omitting notice of proposed rule making and public procedure thereon as unnecessary and impracticable and for making the interim statement of policy effective upon publication in the FEDERAL REGISTER without the customary 30-day notice.

Accordingly, pursuant to NEPA, FWPCA, the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following interim statement of policy is published as a document subject to codification, to be effective on January 29, 1973.

The Commission invites all interested persons who desire to submit written comments or suggestions for consideration in connection with the statement to send them to the Secretary of the Commission, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Staff, on or before March 15, 1973. Consideration will be given to such submissions with the view to possible further amendments. The Commission expects in any event to make conforming changes to the language in 10 CFR Part 50, Appendix D, in the near future after March 15, 1973. Copies of comments received by the Commission may be examined at the Commission's Public Document Room, 1717 H Street NW., Washington, DC.

INTERIM POLICY STATEMENT ON IMPLEMENTATION OF SECTION 511 OF THE FEDERAL WATER POLLUTION CONTROL ACT AMENDMENTS OF 1972 (FWPCA)

1. *Applicability.* This statement is effective immediately and shall apply to all licensing proceedings subject to 10 CFR Part 50, Appendix D, involving facilities or activities which may result in the discharge of a pollutant into the navigable waters, as defined in section 502(12)(A) of the FWPCA, and as to which no final Commission action had been taken prior to enactment of the FWPCA.

2. *Definition of terms.* As used in this statement:

a. Limitations or other requirements promulgated or imposed pursuant to the FWPCA means effluent limitations or other requirements promulgated or imposed pursuant to sections 208(e), 301, 302, 303(e), 304(b), 306, 307, 316, 318, 401, 402, 403, or 404 of the FWPCA. It also includes (i) Water quality standards continued in effect or promulgated pursuant to sections 303(a), 303(b), or 303(c) of the FWPCA; (ii) maximum daily loads for pollutants and maximum daily thermal loads, promulgated pursuant to section 303(d) of the FWPCA; and (iii) limitations or other requirements of State law under authority preserved by section 510 of the FWPCA, but only if and to the extent that such limitations or other requirements covered by this subpart (iii) are imposed and set forth in a certification pursuant to section 401(d) of the FWPCA or are imposed and set forth as a condition in a permit issued pursuant to section 402 of the FWPCA. It does not include (i) effluent limitations or other requirements regarding source, byproduct or special nuclear materials, which are subject to regulation by the Commission pursuant to the Atomic Energy Act of 1954, as amended, or (ii) limitations or other requirements promulgated or imposed pursuant to any other Federal law.

b. Pollutant discharge system means equipment or a mode of operation or procedure designed or intended for the control of discharge of pollutants, as that last phrase is defined in section 502(12) of the FWPCA. It does not include equipment or mode of operation or procedure designed or intended for the control of source, byproduct or special nuclear materials, which are subject to regulation by the Commission pursuant to the Atomic Energy Act of 1954, as amended.

c. Cooling water intake structure location, design, construction, and capacity means cooling water intake structure location, design, construction, and capacity within the meaning of section 316(b) of the FWPCA.

3. *Authority to impose requirements or limitations pursuant to National Environmental Policy Act of 1969 (NEPA).* If and to the extent that there are applicable limitations or other requirements promulgated or imposed pursuant to the FWPCA, different limitations or requirements will not be imposed by the Commission pursuant to NEPA as a condition to any permit or license, provided however, that limitations or other requirements of State law under authority preserved by section 510 of the FWPCA which are imposed and set forth in a certification pursuant to section 401(d) of the FWPCA or imposed and set forth as a condition in a permit issued pursuant to section 402 of the FWPCA, shall be regarded as only minimum limitations or requirements and the Commission shall retain any authority under NEPA to impose more stringent limitations or requirements.

4. *Alternatives.* a. Neither alternative cool-

ing water intake structure location, design, construction, and capacity, nor alternative pollutant discharge systems will be considered by the Commission pursuant to NEPA (i) if a permit has been received for the facility or activity pursuant to section 402 of the FWPCA and a detailed statement with respect to issuance of that permit has been prepared pursuant to section 102(2)(C) of NEPA, or (ii) if and to the extent that conditions imposed as a part of the license or permit for the facility or activity pursuant to section 401(d) of the FWPCA require that a particular alternative be adopted, or (iii) if and to the extent that a permit or determination with a condition requiring the adoption of a particular alternative has been issued for the facility or activity pursuant to sections 208(b)(2)(C)(ii) and 303(e)(3)(B), 318, 402 or 404 of the FWPCA.

b. Alternative pollutant discharge systems will not be considered by the Commission pursuant to NEPA where effluent limitations have been imposed on the facility or activity under sections 301(c) or 302 of the FWPCA.

c. Neither alternative sites, facilities, or activities, nor alternative systems will be considered by the Commission pursuant to NEPA if and to the extent that a determination made with respect to the facility or activity under sections 208(b)(2)(C)(ii) and 303(e)(3)(B) of the FWPCA requires as a condition that a particular site, facility, or activity, or system be adopted.

d. To the maximum extent practicable any alternatives considered by the Commission pursuant to NEPA shall be considered by following procedures similar to those described in paragraph 5.

5. *Cost-benefit balances.* a. Except as provided in paragraphs b. and c., if and to the extent that there are applicable limitations or other requirements promulgated or imposed pursuant to the FWPCA, in considering the costs and benefits of a proposed action pursuant to NEPA, the Commission will determine whether the facility or activity that is the subject of the licensing action will comply with such limitations or other requirements.

(1) If it is determined that the facility or activity, or any part thereof, will not comply with such limitations or other requirements, then the facility or activity, or particular part in question, shall not be approved in the AEC license or permit.

(2) If it is determined that the facility or activity will comply with such limitations or other requirements, then the Commission will evaluate environmental impact on the basis of discharges or other activities associated with the facility or activity to be licensed which are at the level of such limitations or other requirements.

In making a determination in regard to compliance, as provided hereinabove, AEC will give due regard to the views on this matter of the Environmental Protection Agency and, where appropriate, of cognizant State and interstate agencies which exercise authority derived from the FWPCA.

b. Where limitations or other requirements of State law under authority preserved by section 510 of the FWPCA are imposed and set forth in a certification under section 401(d) of the FWPCA or are imposed and set forth as a condition in a permit issued pursuant to section 402 of the FWPCA, the Commission will, in considering costs and benefits of a proposed action pursuant to NEPA, evaluate environmental impact on the basis of discharges or other activities associated with the facility or activity which are in compliance with said limitations or other requirements. In considering the costs and benefits of a proposed action pursuant to NEPA the Commission will accord due consideration to the fact that the facility or activity, or part thereof, will meet limitations or other requirements more stringent than such limitations or other requirements of State law by evaluating and giving consideration to environmental impact of the facility or activity accordingly.

c. (1) The Commission will not determine whether applicable limitations or other requirements promulgated or imposed pursuant to the FWPCA will be complied with if and to the extent that such a determination has been made (i) under sections 208(b)(2)(C)(ii) and 303(e)(3)(B), or (ii) sections 301(c), 302, 318, 401, or 402, or (iii) section 404 of the FWPCA. In such cases, the Commission will accept the determination made under these provisions, *Provided, however,* That the Commission will determine whether applicable limitations or other requirements promulgated or imposed pursuant to the FWPCA will be complied with notwithstanding that a determination has been made under section 401 of the FWPCA where there has been provided a certification that there is not an applicable limitation under sections 301(b) and 302 of the FWPCA and there is not an applicable standard under sections 306 and 307 of the FWPCA.

6. *Effect on Appendix D.* To the extent that there is a conflict between any of the provisions of this interim statement of policy and the provisions of 10 CFR Part 50, Appendix D, the provisions of this statement shall govern.

Published January 29, 1973

PROTECTION AGAINST ACCIDENTS IN NUCLEAR POWER REACTORS

Interim General Statement of Policy

A paramount objective of the Atomic Energy Commission in regulating the design, construction and operation of nuclear power plants is the protection of the public health and safety. Although the operation of nuclear power plants is not completely risk-free, the safety objective of the AEC is and always has been to assure that the risk from normal operation and postulated accidents is maintained at an acceptably low level and to assure that the likelihood of more severe accidents is extremely small.

The Commission's safety regulations set forth a comprehensive three-level approach to meet this objective. First, nuclear power plants are required to be designed and constructed with a high degree of reliability so that failures or malfunctions that could lead to accidents are highly improbable. An essential part of this first level of safety is the requirement for a comprehensive quality assurance program for plant design, construction, and operation. The second level of safety is the required provision for measures to forestall or cope with incidents and malfunctions that could occur notwithstanding the assurance offered by careful plant design, construction, and operation. For example, plants are required to be equipped with reactor protection systems to terminate the nuclear chain reaction quickly and reliably if plant conditions should require such action, and provision is made for leak detection systems to provide indication of incipient fuel cladding failures or degradation of the reactor coolant system pressure boundary well before leaks become safety problems.

The third level of safety is unique to nuclear power plants. A series of highly unlikely major failures of plant components is postulated as a set of design basis accidents, and safety systems are required to be installed to control all such postulated events. An example of such a postulated failure is the loss-of-coolant accident used as a design basis for light water power reactors; emergency core cooling systems, whose requirements were recently strengthened in revised regulations (39 FR 1001, January 4, 1974), and containment are provided to mitigate the consequences of such accidents. The Commission's regulations in 10 CFR Parts 50, "Licensing of Production and Utilization Facilities", and 100, "Reactor Site Criteria", are complementary elements of this third level of safety. Part 100 requires, in effect, that stationary nuclear power reactors be so designed that no design basis accident will result in calculated offsite doses exceeding specified guideline values. These guideline values are well below levels at which serious injury or death would be expected to occur.

A comprehensive evaluation of the level of risk associated with postulated accidents in large stationary light water nuclear power reactors has been underway on behalf of the AEC under the di-

rection of Professor Norman Rasmussen, professor of nuclear engineering at the Massachusetts Institute of Technology. This study, entitled "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, has been released in draft form for comment by interested persons. The draft study is the result of over two years of effort by a large multi-disciplinary team of experts, and promises to provide the most definitive evaluation to date of the risks due to many unlikely accidents in large light water power reactors. The study will be issued in final form after evaluation of the comments received on the draft study.

This draft study deserves the most careful consideration by all individuals, both inside and outside the Commission, who have an interest in nuclear power reactor safety. The study when completed will be the subject of thorough evaluation by the Commission, the independent Advisory Committee on Reactor Safeguards, and the Commission's staff with respect to both the basic question whether the risks portrayed by the study are acceptable from the standpoint of the Commission's statutory responsibility to protect the health and safety of the public, and the related question whether any changes in the Commission's safety or environmental regulations are warranted. Because of the importance of the study and the Commission's desire for early public comment as to its content and implications, the study has been made available for public review and comment in this draft form. For the interim period, the Commission believes that the general public and the nuclear industry should be apprised of the Commission's present position on the basis of its current evaluation of the draft study—namely, whether, in the Commission's view, any immediate or short-term action is called for as a result of the draft study.

The draft study indicates that the risk from the operation of light water nuclear power reactors is very low. The draft study, while employing a different methodological approach from that reflected in 10 CFR Part 100, indicates that the approach to safety as set forth in the Commission's regulations, including Parts 50 and 100, has been successful and in accord with the Commission's requirement that there be reasonable assurance that nuclear power reactors can be operated without endangering the health and safety of the public.

In the approach to safety reflected in the Commission's regulations, postulated accidents, for purposes of analysis, are divided into two categories—"credible" and "incredible". The former ("credible") are considered to be within the category of design basis accidents. Protective measures are required and provided for all those postulated accidents falling within that category, and proposed sites are evaluated by taking into account the conservatively calculated consequences of a spectrum of severe postulated accidents. Those accidents falling within the "incredible" category are considered to be so improbable that no such protective measures are required.

By contrast, all conceivable accidents are considered in the draft study, and each type of accident is characterized as to both probability and consequences.

The application of Parts 50 and 100 helps assure public safety by basing protection to the public, both in design and siting, on a very conservative standard for determining and calculating the consequences of postulated potentially severe "credible" accidents. Part 100 was promulgated at a time when neither the probabilities nor the consequences of these accidents could be calculated with the desired degree of precision. It was, therefore, considered prudent, as a compensating measure, to require that the consequences of these potentially serious accidents be calculated very conservatively. Similarly, the acceptable dose guideline values in Part 100 used for evaluating the suitability of reactor sites were set conservatively low to compensate for any uncertainties in accident analysis. The draft study, on the other hand, offers a methodology for calculating, as realistically as is now possible, both the probabilities and the consequences of a wide spectrum of accidents, including those considered "incredible" for purposes of site evaluation under Part 100. Stated in another manner, under the approach taken in the draft study the distinction recognized in Part 100 between "credible" and "incredible" events is eliminated; instead, probability distribution functions for various accident consequences are provided.

Notwithstanding the differences in methodology between Parts 50 and 100 and the draft study, it is clear from the study that the safety and environmental risks from accidents in nuclear power reactors are deemed far lower than the risks from other natural and man-caused events.

The second question—whether any changes in the Commission's safety or environmental regulations are warranted at this time—also requires consideration. It is the Commission's present view that changes in its existing licensing regulations on the basis of the draft study would be premature until critical comment on the study has been received and analyzed and the final study has been completed and evaluated. Pending this, conservatism requires that neither the lower level consequences nor the lower level risks indicated in the draft study be used as a basis for relaxing the licensing requirements of Parts 50 and 100.

Similarly, discrete parts of the draft study cannot be considered in isolation from the study's overall risk assessment as a basis for present regulatory change. Thus, while the draft study indicates that the probabilities of a limited number of postulated core melt-down accidents may be appreciably higher than previously estimated, the probable consequences of such accidents are far less severe than previously expected and, more importantly, the risks (compounded of probabilities and consequences) from such accidents are very low—lower in fact than had been previously expected. In this connection, it is pertinent to note the reservations of the authors of the draft study with respect to the inappropriateness of direct application of the study's results and methods to licensing decisions and other purposes outside its intended purpose of risk evaluation.

Accordingly, it is the interim position of the Commission that, pending completion and detailed evaluation of the final study, including public comment thereon, (1) no changes in the Commission's safety or environmental regulations pertaining to nuclear power plants are now warranted, (2) the Commission's existing requirements should not be relaxed, and (3) the contents of the draft study are not an appropriate basis for licensing decisions. In reaching this interim position, the Commission considers that neither consequences nor probabilities should be considered alone; that conservatism requires that neither the lower level consequences nor lower level risks indicated in the draft study should be used as a basis at this time for relaxing the Commission's existing requirements; and that the very low resultant risk described in the draft study amply justifies the conclusion that no immediate action is required or appropriate as a result of the draft study's present assessment of the probabilities and consequences of core melt-down. The foregoing position is not intended to preclude the possibility of later interim changes in the Commission's regulations during the period in which both the draft and final studies are under consideration if the Commission determines that such changes are warranted, either on the basis of its own review or as a result of public comment.

All interested persons who desire to submit written comments or suggestions for consideration in connection with this interim general statement of policy or the draft study discussed herein, including any comments or suggestions as to matters considered appropriate for rule making, should send them to the Secretary of the Commission, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Public Proceedings Staff, on or before November 1, 1974. Copies of comments received, as well as copies of the draft study "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, may be examined at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C.

Effective date: The foregoing interim general statement of policy is effective August 27, 1974.

Published August 27, 1974

39 FR 30964
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PART 20 STANDARDS FOR PROTECTION AGAINST RADIATION

GENERAL PROVISIONS

PERMISSIBLE DOSES, LEVELS, AND CONCENTRATIONS

PRECAUTIONARY PROCEDURES

WASTE DISPOSAL

RECORDS, REPORTS, AND NOTIFICATION

EXCEPTIONS AND ADDITIONAL REQUIREMENTS

ENFORCEMENT

AUTHORITY: The provisions of this Part 20 issued under secs. 53, 63, 65, 81, 103, 104, 161, 68 Stat. 930, 933, 935, 936, 937, 948, as amended; 42 U.S.C. 2073, 2093, 2095, 2111, 2133, 2134, 2201. For the purposes of sec. 223, 68 Stat. 958, as amended; 42 U.S.C. 2273, §§20.401-20.409, issued under sec. 161 o., 68 Stat. 950, as amended; 42 U.S.C. 2201 (o). Secs. 202, 206, Pub. L. 93-438, 88 Stat. 1244, 1246 (42 U.S.C. 5842, 5846).

§ 20.1 Purpose.

(a) The regulations in this part establish standards for protection against radiation hazards arising out of activities under licenses issued by the Nuclear Regulatory Commission and are issued pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974. 40 FR 8774

(b) The use of radioactive material or other sources of radiation not licensed by the Commission is not subject to the regulations in this part. However, it is the purpose of the regulations in this part to control the possession, use, and transfer of licensed material by any licensee in such a manner that exposure to such material and to radiation from such material, when added to exposures to unlicensed radioactive material and to other unlicensed sources of radiation in the possession of the licensee, and to radiation therefrom, does not exceed the standards of radiation protection prescribed in the regulations in this part. 25 FR 10914

(c) In accordance with recommendations of the Federal Radiation Council, approved by the President, persons engaged in activities under licenses issued by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974 should, in addition to complying with the requirements set forth in this part, make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as far below the limits specified in this part as practicable. The term "as far below the limits specified in this part as practicable" means as low as is practicably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and in relation to the utilization of atomic energy in the public interest. 35 FR 18385

§ 20.2 Scope.

The regulations in this part apply to all persons who receive, possess, use, or transfer material licensed pursuant to the regulations in Parts 30 through 35, 40, or 70 of this chapter, including persons licensed to operate a production or utilization facility pursuant to Part 50 of this chapter. 40 FR 8774

§ 20.3 Definitions.

(a) As used in this part:

(1) "Act" means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto;

(2) "Airborne radioactive material" means any radioactive material dispersed in the air in the form of dusts, fumes, mists, vapors, or gases;

(3) "Byproduct 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;

(4) "Calendar quarter" means not less than 12 consecutive weeks nor more than 14 consecutive weeks. The first calendar quarter of each year shall begin in January and subsequent calendar quarters shall be such that no day is included in more than one calendar quarter or omitted from inclusion within a calendar quarter. No licensee shall change the method observed by him of determining calendar quarters except at the beginning of a calendar year.

(5) "Commission" means the Nuclear Regulatory Commission or its duly authorized representatives;

(6) "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;

(7) "Individual" means any human being;

(8) "Licensed material" means source material, special nuclear material, or byproduct material received, possessed, used, or transferred under a general or specific license issued by the Commission pursuant to the regulations in this chapter;

(9) "License" means a license issued under the regulations in Part 30, 40, or 70 of this chapter. "Licensee" means the holder of such license;

(10) "Occupational dose" includes exposure of an individual to radiation (i) in a restricted area; or (ii) in the course of employment in which the individual's duties involve exposure to radiation; provided, that "occupational dose" shall not be deemed to include any exposure of an individual to radiation for the purpose of medical diagnosis or medical therapy of such individual.

(11) "Person" means (i) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, Government agency other than the Commission or the Administration (except that the Administration shall be considered a person within the meaning of the regulations in this part to the extent that its facilities and activities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974 (88 Stat. 1244)), any State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (ii) any legal successor, representative, agent, or agency of the foregoing.

(12) "Radiation" means any or all of the following: alpha rays, beta rays, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, and other atomic particles; but not sound or radio waves, or visible, infrared, or ultraviolet light;

(13) "Radioactive material" includes any such material whether or not subject to licensing control by the Commission;

(14) "Restricted area" means any area access to which is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. "Restricted area" shall not include any areas used as residential quarters, although a separate room or rooms in a residential building may be set apart as a restricted area;

(15) "Source material" means (i) uranium or thorium, or any combination thereof, in any physical or chemical form: or (ii) ores which contain by weight one-twentieth of one percent (0.05%) or more of *a.* uranium, *b.* thorium or *c.* any combination thereof. Source material does not include special nuclear material.

(16) "Special nuclear material" means (i) 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 (ii) any material artificially enriched by any of the foregoing but does not include source material;

(17) "Unrestricted area" means any area access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

(18) "Administration" means the Energy Research and Development Administration or its duly authorized representatives.

(b) Definitions of certain other words and phrases as used in this part are set forth in other sections, including:

(1) "Airborne radioactivity area" defined in § 20.203;

(2) "Radiation area" and "high radiation area" defined in § 20.202;

(3) "Personnel monitoring equipment" defined in § 20.202;

(4) "Survey" defined in § 20.201;

(5) Units of measurement of dose (rad, rem) defined in § 20.4;

(6) Units of measurement of radioactivity defined in § 20.5.

§ 20.4 Units of radiation dose.

(a) "Dose," as used in this part, is the quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body. When the regulations in this part specify a dose during a period of time, the dose means the total quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body during such period of time. Several different units of dose are in current use. Definitions of units as used in this part are set forth in paragraphs (b) and (c) of this section.

(b) The rad, as used in this part, is a measure of the dose of any ionizing radiation to body tissues in terms of the energy absorbed per unit mass of the tissue. One rad is the dose corresponding to the absorption of 100 ergs per gram of tissue. (One millirad (mrad)=0.001 rad.)

(c) The rem, as used in this part, is a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one roentgen (r) of X-rays. (One millirem (mrem)=0.001 rem.) The relation of the rem to other dose units depends upon the biological effect under consideration and upon the conditions of irradiation. For the purpose of the regulations in this part, any of the following is considered to be equivalent to a dose of one rem:

(1) A dose of 1 r due to X— or gamma radiation;

(2) A dose of 1 rad due to X—, gamma, or beta radiation;

(3) A dose of 0.1 rad due to neutrons or high energy protons;

(4) A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye;

If it is more convenient to measure the neutron flux, or equivalent, than to determine the neutron dose in rads, as provided in subparagraph (3) of this paragraph, one rem of neutron radiation may, for purposes of the regulations in this part, be assumed to be equivalent to 14 million neutrons per square centimeter incident upon the body; or, if there exists sufficient information to estimate with reasonable accuracy the approximate distribution in energy of the neutrons, the incident number of neutrons per square centimeter equivalent to one rem may be estimated from the following table:

NEUTRON FLUX DOSE-EQUIVALENTS

Neutron energy (Mev)	Number of neutrons per square centimeter equivalent to a dose of 1 rem (neutrons/cm²)	Average flux to deliver 100 millirem in 40 hours (neutrons/cm² per sec.)
Thermal	970×10^{6}	670
0.0001	720×10^{6}	500
0.005	820×10^{6}	570
0.02	400×10^{6}	280
0.1	120×10^{6}	80
0.5	43×10^{6}	30
1.0	26×10^{6}	18
2.5	29×10^{6}	20
5.0	26×10^{6}	18
7.5	24×10^{6}	17
10	24×10^{6}	17
10 to 30	14×10^{6}	10

(d) For determining exposures to X or gamma rays up to 3 Mev, the dose limits specified in §§ 20.101 to 20.104, inclusive, may be assumed to be equivalent to the "air dose". For the purpose of this part "air dose" means that the dose is measured by a properly calibrated appropriate instrument in air at or near the body surface in the region of highest dosage rate.

§ 20.5 Units of radioactivity.

(a) Radioactivity is commonly, and for purposes of the regulations in this part shall be, measured in terms of disintegrations per unit time or in curies. One curie=3.7×10^{10} disintegrations per second (dps)=2.2×10^{12} disintegrations per minute (dpm). Commonly used submultiples of the curie are the millicurie and the microcurie:

(1) One millicurie (mCi) [1]=0.001 curie (Ci) [1]=3.7×10^{7} dps.

(2) One microcurie (μCi) [1]=0.000001 curie=3.7×10^{4} dps.

(b) For purposes of the regulations in this part, it may be assumed that the

[1] Wherever possible, the appropriate unit should be written out as "curie(s)," "millicurie(s)," or "microcurie(s)," and the abbreviations should not be used.

daughter activity concentrations in the following table are equivalent to an air concentration of 10^{-7} microcuries of Radon 222 per milliliter of air in equilibrium with the daughters RaA, RaB, RaC, and RaC':

Maximum time between collection and measurement (hours) [8]	Alpha-emitting daughter activity collected per milliliter of air	
	Microcuries/cc	Total alpha disintegrations per minute per cc.
0.5	7.2×10^{-5}	0.16
1	4.5×10^{-5}	0.10
2	1.3×10^{-5}	0.028
3	0.3×10^{-5}	0.0099

(c) *

§ 20.6 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.

§ 20.7 Communications.

Except where otherwise specified in this part, all communications and reports concerning the regulations in this part should be addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street NW., Washington, D.C.; or at 7920 Norfolk Avenue, Bethesda, Maryland.

40 FR 8774

** PERMISSIBLE DOSES, LEVELS, AND CONCENTRATIONS

§ 20.101 Exposure of individuals to radiation in restricted areas.

(a) Except as provided in paragraph (b) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of the limits specified in the following table:

25 FR 10914

Rems per calendar quarter

1. Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads 1¼
2. Hands and forearms; feet and ankles 18¾
3. Skin of whole body 7½

(b) A licensee may permit an individual in a restricted area to receive a dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) During any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems; and

(2) The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N–18) rems where "N" equals the individual's age in years at his last birthday; and

(3) The licensee has determined the individual's accumulated occupational dose to the whole body on Form NRC-4, or on a clear and legible record containing all the information required in that form; and has otherwise complied with the requirements of § 20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active blood-forming organs, head and trunk, or lens of eye.

§ 20.102 Determination of accumulated dose.

(a) This section contains requirements which must be satisfied by licensees who propose, pursuant to paragraph (b) of § 20.101, to permit individuals in a restricted area to receive exposure to radiation in excess of the limits specified in paragraph (a) of § 20.101.

(b) Before permitting any individual in a restricted area to receive exposure to radiation in excess of the limits specified in paragraph (a) of § 20.101, each licensee shall:

25 FR 10914

(1) Obtain a certificate on Form NRC-4, or on a clear and legible record containing all the information required in that form, signed by the individual showing each period of time after the individual attained the age of 18 in which the individual received an occupational dose of radiation; and

(2) Calculate on Form NRC-4 in accordance with the instructions appearing therein, or on a clear and legible record containing all the information required in that form, the previously accumulated occupational dose received by the individual and the additional dose allowed for that individual under § 20.101(b).

(c)(1) In the preparation of Form NRC-4, or a clear and legible record containing all the information required in that form, the licensee shall make a reasonable effort to obtain reports of the individual's previously accumulated occupational dose. For each period for which the licensee obtains such reports, the licensee shall use the dose shown in the report in preparing the form. In any case where a licensee is unable to obtain reports of the individual's occupational dose for a previous complete calendar quarter, it shall be assumed that the individual has received the occupational dose specified in whichever of the following columns apply:

Part of body	Column 1 Assumed exposure in rems for calendar quarters prior to Jan. 1, 1961	Column 2 Assumed exposure in rems for calendar quarters beginning on or after Jan. 1, 1961
Whole body, gonads, active blood-forming organs, head and trunk, lens of eye.	3¾	1¼

(2) The licensee shall retain and preserve records used in preparing Form NRC-4.

If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date, as specified in paragraph (b) of § 20.101, the excess may be disregarded.

§ 20.103 Exposure of individuals to concentrations of radioactive material in restricted areas.

(a) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual in a restricted area to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table I, of this part. "Expose" as used in this section means that the individual is present in an airborne concentration. No allowance shall be made for the use of protective clothing or equipment, or particle size, except as authorized by the Commission pursuant to paragraph (c) of this section.

(b) The limits given in Appendix B, Table I, of this part are based upon exposure to the concentrations specified for forty hours in any period of seven consecutive days. In any such period where the number of hours of exposure is less than forty, the limits specified in the table may be increased proportionately. In any such period where the number of hours of exposure is greater than forty, the limits specified in the table shall be decreased proportionately.

(c)(1) Except as authorized by the Commission pursuant to this paragraph, no allowance shall be made for particle size or the use of protective clothing or equipment in determining whether an individual is exposed to an airborne concentration in excess of the limits specified in Appendix B, Table I.

(2) The Commission may authorize a licensee to expose an individual in a restricted area to airborne concentrations in excess of the limits specified in Appendix B, Table I, upon receipt of an application demonstrating that the concentration is composed in whole or in part of particles of such size that such particles are not respirable; and that the individual will not inhale the concentrations in excess of the limits established in Appendix B, Table I. Each application under this subparagraph shall include an analysis of particle sizes in the concentrations; and a description

[8] The duration of sample collection and the duration of measurement should be sufficiently short compared to the time between collection and measurement, as not to have a statistically significant effect upon the results.

* Deleted 39 FR 23990.

** Amended 36 FR 1466.

of the methods used in determining the particle sizes.

(3) The Commission may authorize a licensee to expose an individual in a restricted area to airborne concentrations in excess of the limits specified in Appendix B, Table I, upon receipt of an application demonstrating that the individual will wear appropriate protective equipment and that the individual will not inhale, ingest or absorb quantities of radioactive material in excess of those which might otherwise be permitted under this part for employees in restricted areas during a 40-hour week. Each application under this subparagraph shall contain the following information:

(i) A description of the protective equipment to be employed, including the efficiency of the equipment for the material involved;

(ii) Procedures for the fitting, maintenance and cleaning of the protective equipment; and

(iii) Procedures governing the use of the protective equipment, including supervisory procedures and length of time the equipment will be used by the individuals in each work week. The proposed periods for use of the equipment by any individual should not be of such duration as would discourage observance by the individual of the proposed procedures; and

(iv) The average concentrations present in the areas occupied by employees.

25 FR 10914

§ 20.104 Exposure of minors.

(a) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age, to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of 10 percent of the limits specified in the table in paragraph (a) of § 20.101.

(b) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area, who is under 18 years of age to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than a week.

(c) The provisions of paragraph (c) of § 20.103, shall apply to exposures subject to paragraph (b) of this section.

§ 20.105 Permissible levels of radiation in unrestricted areas.

(a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

25 FR 10914

(1) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour, or

(2) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

§ 20.106 Radioactivity in effluents to unrestricted areas.

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B", Table II of this part, except as authorized pursuant to § 20.302 or paragraph (b) of this section. For purposes of this section concentrations may be averaged over a period not greater than one year.

29 FR 14434

(b) An application for a license or amendment may include proposed limits higher than those specified in paragraph (a) of this section. The Commission will approve the proposed limits if the applicant demonstrates:

(1) That the applicant has made a reasonable effort to minimize the radioactivity contained in effluents to unrestricted areas; and

(2) That it is not likely that radioactive material discharged in the effluent would result in the exposure of an individual to concentrations of radioactive material in air or water exceeding the limits specified in Appendix "B", Table II of this part.

29 FR 14434

(c) An application for higher limits pursuant to paragraph (b) of this section shall include information demonstrating that the applicant has made a reasonable effort to minimize the radioactivity discharged in effluents to unrestricted areas, and shall include, as pertinent:

(1) Information as to flow rates, total volume of effluent, peak concentration of each radionuclide in the effluent, and concentration of each radionuclide in the effluent averaged over a period of one year at the point where the effluent leaves a stack, tube, pipe, or similar conduit;

(2) A description of the properties of the effluents, including:

(i) chemical composition;

(ii) physical characteristics, including suspended solids content in liquid effluents, and nature of gas or aerosol for air effluents;

(iii) the hydrogen ion concentrations (p^H) of liquid effluents; and

(iv) the size range of particulates in effluents released into air.

(3) A description of the anticipated human occupancy in the unrestricted area where the highest concentration of radioactive material from the effluent is expected, and, in the case of a river or stream, a description of water uses downstream from the point of release of the effluent.

(4) Information as to the highest concentration of each radionuclide in an unrestricted area, including anticipated concentrations averaged over a period of one year:

(i) In air at any point of human occupancy; or

(ii) In water at points of use downstream from the point of release of the effluent.

(5) The background concentration of radionuclides in the receiving river or stream prior to the release of liquid effluent.

(6) A description of the environmental monitoring equipment, including sensitivity of the system, and procedures and calculations to determine concentrations of radionuclides in the unrestricted area and possible reconcentrations of radionuclides.

(7) A description of the waste treatment facilities and procedures used to reduce the concentration of radionuclides in effluents prior to their release.

(d) For the purposes of this section the concentration limits in Appendix "B", Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe or similar conduit may be determined with respect to the point where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion, or decay between the point of discharge and the boundary.

(e) In addition to limiting concentrations in effluent streams, the Commission may limit quantities of radioactive materials released in air or water during a specified period of time if it appears that the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, averaged over a period not exceeding one year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentration of radioactive materials specified in Appendix "B", Table II of this part.

(f) The provisions of this section do not apply to disposal of radioactive material into sanitary sewerage systems, which is governed by § 20.303

§ 20.107 Medical diagnosis and therapy.

Nothing in the regulations in this part shall be interpreted as limiting the intentional exposure of patients to radiation for the purpose of medical diagnosis or medical therapy.

25 FR 10914

§ 20.108 Orders requiring furnishing of bio-assay services.

Where necessary or desirable in order to aid in determining the extent of an

individual's exposure to concentrations of radioactive material, the Commission may incorporate appropriate provisions in any license, directing the licensee to make available to the individual appropriate bio-assay services and to furnish a copy of the reports of such services to the Commission.

PRECAUTIONARY PROCEDURES

§ 20.201 Surveys.

(a) As used in the regulations in this part, "survey" means an evaluation of the radiation hazards incident to the production, use, release, disposal, or presence of radioactive materials or other sources of radiation under a specific set of conditions. When appropriate, such evaluation includes a physical survey of the location of materials and equipment, and measurements of levels of radiation or concentrations of radioactive material present.

(b) Each licensee shall make or cause to be made such surveys as may be necessary for him to comply with the regulations in this part.

§ 20.202 Personnel monitoring.

(a) Each licensee shall supply appropriate personnel monitoring equipment to, and shall require the use of such equipment by:

(1) Each individual who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 25 percent of the applicable value specified in paragraph (a) of § 20.101.

(2) Each individual under 18 years of age who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 5 percent of the applicable value specified in paragraph (a) of § 20.101.

(3) Each individual who enters a high radiation area.

(b) As used in this part,

(1) "Personnel monitoring equipment" means devices designed to be worn or carried by an individual for the purpose of measuring the dose received (e. g., film badges, pocket chambers, pocket dosimeters, film rings, etc.);

(2) "Radiation area" means any area, accessible to personnel, in which there exists radiation, originating in whole or in part within licensed material, at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 100 millirems;

(3) "High radiation area" means any area, accessible to personnel, in which there exists radiation originating in whole or in part within licensed material at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 millirem.

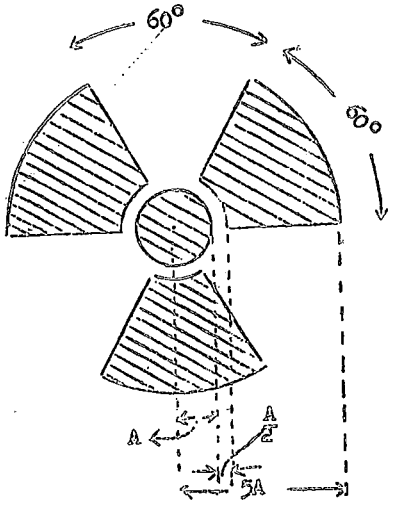
35 FR 5033

§ 20.203 Caution signs, labels, signals, and controls.

(a) *General.* (1) Except as otherwise authorized by the Commission, symbols prescribed by this section shall use the conventional radiation caution colors (magenta or purple on yellow background). The symbol prescribed by this section is the conventional three-bladed design:

RADIATION SYMBOL

1. Cross-hatched area is to be magenta or purple.
2. Background is to be yellow.



(2) In addition to the contents of signs and labels prescribed in this section, licensees may provide on or near such signs and labels any additional information which may be appropriate in aiding individuals to minimize exposure to radiation or to radioactive material.

(b) *Radiation areas.* Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION [1]
RADIATION AREA

(c) *High radiation areas.* (1) Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION [1]
HIGH RADIATION AREA

(2) Each entrance or access point to a high radiation area shall be:

(i) Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirems in 1 hour upon entry into the area; or

(ii) Equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry; or

(iii) Maintained locked except during periods when access to the area is required, with positive control over each individual entry.

(3) The controls required by subparagraph (2) of this paragraph shall be established in such a way that no individual will be prevented from leaving a high radiation area.

(4) In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for the controls required by subparagraph (2) of this paragraph.

(5) Any licensee, or applicant for a license, may apply to the Commission for approval of methods not included in subparagraphs (2) and (4) of this paragraph for controlling access to high radiation areas. The Commission will approve the proposed alternatives if the licensee or applicant demonstrates that the alternative methods of control will prevent unauthorized entry into a high radiation area, and that the requirement of subparagraph (3) of this paragraph is met.

35 FR 5033

(d) *Airborne radioactivity areas.* (1) As used in the regulations in this part, "airborne radioactivity area" means (i) any room, enclosure, or operating area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the amounts specified in Appendix B, Table I, Column 1 of this part; or (ii) any room, enclosure, or operating area in which airborne radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in Appendix B, Table I, Column 1 of this part.

(2) Each airborne radioactivity area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION [1]
AIRBORNE RADIOACTIVITY AREA

(e) *Additional requirements.* (1) Each area or room in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding 10 times the quantity of such material specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION [1]
RADIOACTIVE MATERIAL(S)

(2) Each area or room in which natural uranium or thorium is used or stored in an amount exceeding one-hundred times the quantity specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION [1]
RADIOACTIVE MATERIAL(S)

(f) *Containers.* (1) Except as provided in subparagraph (3) of this paragraph, each container of licensed mate-

[1] Or "Danger."

rial shall bear a durable, clearly visible label identifying the radioactive contents.

(2) A label required pursuant to subparagraph (1) of this paragraph shall bear the radiation caution symbol and the words "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". It shall also provide sufficient information[1] to permit individuals handling or using the containers, or working in the vicinity thereof, to take precautions to avoid or minimize exposures.

(3) Notwithstanding the provisions of subparagraph (1) of this paragraph, labeling is not required:

(i) For containers that do not contain licensed materials in quantities greater than the applicable quantities listed in Appendix C of this part.

(ii) For containers containing only natural uranium or thorium in quantities no greater than 10 times the applicable quantities listed in Appendix C of this part.

(iii) For containers that do not contain licensed materials in concentrations greater than the applicable concentrations listed in Column 2, Table I, Appendix B of this part.

(iv) For containers when they are attended by an individual who takes the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established by the regulations in this part.

(v) For containers when they are in transport and packaged and labeled in accordance with regulations of the * Department of Transportation.

(vi) For containers which are accessible[2] only to individuals authorized to handle or use them, or to work in the vicinity thereof, provided that the contents are identified to such individuals by a readily available written record.

(vii) For manufacturing or process equipment, such as nuclear reactors, reactor components, piping, and tanks.

31 FR 10514

§ 20.204 Same: exceptions.

Notwithstanding the provisions of § 20.203,

(a) A room or area is not required to be posted with a caution sign because of the presence of a sealed source provided the radiation level twelve inches from the surface of the source container or housing does not exceed five millirem per hour.

25 FR 10914

(b) Rooms or other areas in hospitals are not required to be posted with caution signs, and control of entrance or access thereto pursuant to § 20.203(c) is not required, because of the presence of patients containing byproduct material provided that there are personnel in attendance who will take the precautions necessary to prevent the exposure of any individual to radiation or radioactive material in excess of the limits established in the regulations in this part.

35 FR 5033

(c) Caution signs are not required to be posted at areas or rooms containing radioactive materials for periods of less than eight hours provided that (1) the materials are constantly attended during such periods by an individual who shall take the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established in the regulations in this part and; (2) such area or room is subject to the licensee's control.

25 FR 10914

(d) A room or other area is not required to be posted with a caution sign, and control is not required for each entrance or access point to a room or other area which is a high radiation area solely because of the presence of radioactive materials prepared for transport and packaged and labeled in accordance with regulations of the Department of Transportation.

35 FR 5033

§ 20.205 Procedures for picking up, receiving, and opening packages.

(a) (1) Each licensee who expects to receive a package containing quantities of radioactive material in excess of the Type A quantities specified in paragraph (b) of this section shall:

(i) If the package is to be delivered to the licensee's facility by the carrier, make arrangements to receive the package when it is offered for delivery by the carrier; or

(ii) If the package is to be picked up by the licensee at the carrier's terminal, make arrangements to receive notification from the carrier of the arrival of the package, at the time of arrival.

(2) Each licensee who picks up a package of radioactive material from a carrier's terminal shall pick up the package expeditiously upon receipt of notification from the carrier of its arrival.

39 FR 17972

(b) (1) Each licensee, upon receipt of a package of radioactive material, shall monitor the external surfaces of the package for radioactive contamination caused by leakage of the radioactive contents, except:

(i) Packages containing no more than the exempt quantity specified in the table in this paragraph;

(ii) Packages containing no more than 10 millicuries of radioactive material consisting solely of tritium, carbon-14, sulfur-35, or iodine-125;

(iii) Packages containing only radioactive material as gases or in special form;

(iv) Packages containing only radioactive material in other than liquid form (including Mo-99/Tc-99m generators) and not exceeding the Type A quantity limit specified in the table in this paragraph; and

(v) Packages containing only radionuclides with half-lives of less than 30 days and a total quantity of no more than 100 millicuries.

The monitoring shall be performed as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or eighteen hours if received after normal working hours.

(2) If removable radioactive contamination in excess of 0.01 microcuries (22,000 disintegrations per minute) per 100 square centimeters of package surface is found on the external surfaces of the package, the licensee shall immediately notify the final delivering carrier and, by telephone and telegraph, the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office shown in Appendix D.

TABLE OF EXEMPT AND TYPE A QUANTITIES

Transport group[1]	Exempt quantity limit (in millicuries)	Type A quantity limit (in curies)
I	.01	0.001
II	0.1	0.020
III	1	3
IV	1	20
V	1	20
VI	1	1000
VII	25,000	1000
Special Form	1	20

39 FR 17972

(c) (1) Each licensee, upon receipt of a package containing quantities of radioactive material in excess of the Type A quantities specified in paragraph (b) of this section, other than those transported by exclusive use vehicle, shall monitor the radiation levels external to the package. The package shall be monitored as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or 18 hours if received after normal working hours.

(2) If radiation levels are found on the external surface of the package in excess of 200 millirem per hour, or at three feet from the external surface of the package in excess of 10 millirem per hour, the licensee shall immediately notify, by telephone and telegraph, the final delivering carrier and the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office shown in Appendix D.

(d) Each licensee shall establish and maintain procedures for safely opening packages in which licensed material is received, and shall assure that such procedures are followed and that due consideration is given to special instructions for the type of package being opened.

38 FR 22220

§ 20.206 Instruction of personnel.

Instructions required for individuals working in or frequenting any portion of a restricted area are specified in § 19.12 of this chapter.

[1] As appropriate, the information will include radiation levels, kinds of material, estimate of activity, date for which activity is estimated, mass enrichment, etc.

[2] For example, containers in locations such as water-filled canals, storage vaults, or hot cells.

* Amended 34 FR 19546.

[1] The definitions of "transport group" and "special form" are specified in § 71.4 of this chapter.

§ 20.207 Storage of licensed materials.

Licensed materials stored in an unrestricted area shall be secured against unauthorized removal from the place of storage.

WASTE DISPOSAL

§ 20.301 General requirement.

No licensee shall dispose of licensed material except:

(a) By transfer to an authorized recipient as provided in the regulations in Part 30, 40, or 70 of this chapter, whichever may be applicable; or

(b) As authorized pursuant to § 20.302; or

(c) As provided in § 20.303 or § 20.304, applicable respectively to the disposal of licensed material by release into sanitary sewerage systems or burial in soil, or in § 20.106 (Radioactivity in Effluents to Unrestricted Areas).

25 FR 10914

§ 20.302 Method for obtaining approval of proposed disposal procedures.

*(a) Any licensee or applicant for a license may apply to the Commission for approval of proposed procedures to dispose of licensed material in a manner not otherwise authorized in the regulations in this chapter. Each application should include a description of the licensed material and any other radioactive material involved, including the quantities and kinds of such material and the levels of radioactivity involved, and the proposed manner and conditions of disposal. The application should also include an analysis and evaluation of pertinent information as to the nature of the environment, including topographical, geological, meteorological, and hydrological characteristics; usage of ground and surface waters in the general area; the nature and location of other potentially affected facilities; and procedures to be observed to minimize the risk of unexpected or hazardous exposures.

26 FR 352

*(b) The Commission will not approve any application for a license to receive licensed material from other persons for disposal on land not owned by the Federal government or by a State government.

36 FR 23138

(c) The Commission will not approve any application for a license for disposal of licensed material at sea unless the applicant shows that sea disposal offers less harm to man or the environment than other practical alternative methods of disposal.

25 FR 10914

§ 20.303 Disposal by release into sanitary sewerage systems.

No licensee shall discharge licensed material into a sanitary sewerage system unless:

(a) It is readily soluble or dispersible in water; and

(b) The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

(1) The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee, will result in an average concentration equal to the limits specified in Appendix B, Table I, Column 2 of this part; or

(2) Ten times the quantity of such material specified in Appendix C of this part; and

(c) The quantity of any licensed or other radioactive material released in any one month, if diluted by the average monthly quantity of water released by the licensee, will not result in an average concentration exceeding the limits specified in Appendix B, Table I, Column 2 of this part; and

(d) The gross quantity of licensed and other radioactive material released into the sewerage system by the licensee does not exceed one curie per year.

Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitations contained in this section.

25 FR 10914

§ 20.304 Disposal by burial in soil.

No licensee shall dispose of licensed material by burial in soil unless:

(a) The total quantity of licensed and other radioactive materials buried at any one location and time does not exceed, at the time of burial, 1,000 times the amount specified in Appendix C of this part; and

(b) Burial is at a minimum depth of four feet; and

(c) Successive burials are separated by distances of at least six feet and not more than 12 burials are made in any year.

§ 20.305 Treatment or disposal by incineration.

No licensee shall treat or dispose of licensed material by incineration except as specifically approved by the Commission pursuant to §§ 20.106(b) and 20.302.

RECORDS, REPORTS, AND NOTIFICATION

§ 20.401 Records of surveys, radiation monitoring, and disposal.

(a) Each licensee shall maintain records showing the radiation exposures of all individuals for whom personnel monitoring is required under § 20.202 of the regulations in this part. Such records shall be kept on Form NRC-5, in accordance with the instructions contained in that form or on clear and legible records containing all the information required by Form NRC-5. The doses entered on the forms or records shall be for periods of time not exceeding one calendar quarter.

39 FR 17972

(b) Each licensee shall maintain records in the same units used in this part, showing the results of surveys required by § 20.201(b), monitoring required by §§ 20.205(b) and 20.205(c), and disposals made under §§ 20.302, 20.303, and 20.304.

35 FR 18110

(c) Records of individual exposure to radiation and to radioactive material which must be maintained pursuant to the provisions of paragraph (a) of this section and records of bio-assays, including results of whole body counting examinations, made pursuant to § 20.108 shall be preserved indefinitely or until the Commission authorizes their disposal. Records which must be maintained pursuant to this part may be maintained in the form of microfilms.

34 FR 7500

§ 20.402 Reports of theft or loss of licensed material.

(a) Each licensee shall report by telephone and telegraph to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D, immediately after its occurrence becomes known to the licensee, any loss or theft of licensed material in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas.

(b) Each licensee who is required to make a telephonic and telegraphic report pursuant to paragraph (a) of this section shall, within 30 days after he learns of the loss or theft, make a report in writing to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D, setting forth the following information:

(1) A description of the licensed material involved, including kind, quantity, chemical, and physical form;

(2) A description of the circumstances under which the loss or theft occurred;

(3) A statement of disposition or probable disposition of the licensed material involved;

(4) Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas;

(5) Actions which have been taken, or will be taken, to recover the material; and

(6) Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

(c) Subsequent to filing the written report the licensee shall also report any substantive additional information on the loss or theft which becomes available to the licensee, within 30 days after he learns of such information.

(d) Any report filed with the Commission pursuant to this section shall be so prepared that names of individuals who may have received exposure to radiation are stated in a separate part of the report.

27 FR 5905

§ 20.403 Notifications of incidents.

(a) *Immediate notification.* Each licensee shall immediately notify the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office

* Redesignated 36 FR 23138.

shown in Appendix D by telephone and telegraph of any incident involving byproduct, source or special nuclear material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $100,000.

(b) *Twenty-four hour notification.* Each licensee shall within 24 hours notify the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D

by telephone and telegraph of any incident involving licensed material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $1,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

§ 20.404 ‡

§ 20.405 Reports of overexposures and excessive levels and concentrations.

(a) In addition to any notification required by § 20.403, each licensee shall make a report in writing within 30 days to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D,

of (1) each exposure of an individual to radiation or concentrations of radioactive material in excess of any applicable limit in this part or in the licensee's license; (2) any incident for which notification is required by § 20.403; and (3) levels of radiation or concentrations of radioactive material (not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any applicable limit set forth in this part or in the licensee's license.

Each report required under this paragraph shall describe the extent of exposure of persons to radiation or to radioactive material, including estimates of each individual's exposure as required by paragraph (b) of this section; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

(b) Any report filed with the Commission pursuant to this section shall include for each individual exposed the name, social security number, and date of birth; and an estimate of the individual's exposure. The report shall be prepared so that this information is stated in a separate part of the report.

(c) †

§ 20.406 †

§ 20.407 Personnel exposure and monitoring reports.

(a) This section applies to each person licensed by the Commission or the Atomic Energy Commission to:

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to § 50.21(b) or § 50.22 of this chapter or a testing facility as defined in § 50.2(r) of this chapter;

(2) Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter;

(3) Possess or use at any one time, for purposes of fuel processing fabrication, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof pursuant to Part 70 of this chapter; or

(4) Possess or use at any one time, for processing or manufacturing for distribution pursuant to Part 30, 32, or 33 of this chapter, byproduct material in quantities exceeding anyone of the following quantities:

Radionuclide [1]	*Quantity in curies*
Cesium-137	1
Cobalt-60	1
Gold-198	100
Iodine-131	1
Iridium-192	10
Krypton-85	1,000
Promethium-147	10
Technetium-99m	1,000

(b) Each person described in paragraph (a) of this section shall, within the first quarter of each calendar year, submit to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, the following reports, applicable to the described licensed activities covering the preceding calendar year: [2]

(1) A report of either (i) the total number of individuals for whom personnel monitoring was required under §§ 20.202(a) or 34.33(a) of this chapter during the calendar year, or (ii) the total number of individuals for whom personnel monitoring was provided during the calendar year; *Provided*, that such total includes at least the number of individuals required to be reported under paragraph (b)(1)(i) of this section. The report shall indicate whether it is submitted in accordance with paragraph (b)(1)(i) or (ii) of this section.

(2) A statistical summary report of the personnel monitoring information recorded by the licensee for individuals for whom personnel monitoring was either required or provided, as described in § 20.407(b)(1), indicating the number of individuals whose total whole body exposure recorded during the previous calendar year was in each of the following estimated exposure ranges:

Estimated Whole Body Exposure Range (Rems) [3]	*Number of individuals in each range*
No measurable exposure	
Measurable exposure less than 0.1	
0.1 to 0.25	
0.25 to 0.5	
0.5 to 0.75	
0.75 to 1	
1 to 2	
2 to 3	
3 to 4	
4 to 5	
5 to 6	
6 to 7	
7 to 8	
8 to 9	
9 to 10	
10 to 11	
11 to 12	
12+	

The low exposure range data are required in order to obtain better information about the exposures actually recorded. This section does not require improved measurements.

§ 20.408 Reports of personnel exposure on termination of employment or work.

When an individual terminates employment with a licensee subject to § 20.407, or an individual assigned to work in such a licensee's facility, but not employed by the licensee, completes his work assignment in the licensee's facility, the licensee shall furnish * to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, a report of the individual's exposure to radiation and radioactive material, incurred during the

‡ Deleted 38 FR 22220.

[1] The Commission may require, as a license condition, or by rule, regulation or order pursuant to § 20.502, reports from licensees who are licensed to use radionuclides not on this list, in quantities sufficient to cause comparable radiation levels.

† Deleted 38 FR 22220.

[2] A licensee whose license expires or terminates prior to, or on the last day of the calendar year, shall submit reports at the expiration or termination of the license, covering that part of the year during which the license was in effect.

[3] Individual values exactly equal to the values separating Exposure Ranges shall be reported in the higher range.

* Amended 38 FR 22220.

34 FR 5254

period of employment or work assignment in the licensee's facility, containing information recorded by the licensee pursuant to §§ 20.401(a) and 20.108. Such report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

38 FR 22220

§ 20.409 Notifications and reports to individuals.

(a) Requirements for notifications and reports to individuals of exposure to radiation or radioactive material are specified in § 19.13 of this chapter.

(b) When a licensee is required pursuant to §§ 20.405 or 20.408 to report to the Commission any exposure of an individual to radiation or radioactive material, the licensee shall also notify the individual. Such notice shall be transmitted at a time not later than the transmittal to the Commission, and shall comply with the provisions of § 19.13(a) of this chapter.

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EXCEPTIONS AND ADDITIONAL REQUIREMENTS

§ 20.501 Applications for exemptions.

The Commission may, upon application by any licensee or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not result in undue hazard to life or property.

§ 20.502 Additional requirements.

The Commission may, by rule, regulation, or order, impose upon any licensee such requirements, in addition to those established in the regulations in this part, as it deems appropriate or necessary to protect health or to minimize danger to life or property.

40 FR 8774

§ 20.601 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, or Title II of the Energy Reorganization Act of 1974, or any regulation or order issued thereunder. A court order may be obtained for the payment of a civil penalty imposed pursuant to section 234 of the Act for violation of section 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Act, or section 206 of the Energy Reorganization Act of 1974, or any rule, regulation, or order issued thereunder, or any term, condition, or limitation of any license issued thereunder, or for any violation for which a license may be revoked under section 186 of the Act. Any person who willfully violates any provision of the Act or any regulation or order issued thereunder may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

APPENDIX A [Reserved]

APPENDIX B

Concentrations in Air and Water Above Natural Background

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Actinium (89)	Ac 227	S	2×10^{-12}	6×10^{-5}	8×10^{-14}	2×10^{-6}
		I	3×10^{-11}	9×10^{-3}	9×10^{-13}	3×10^{-4}
	Ac 228	S	8×10^{-8}	3×10^{-3}	3×10^{-9}	9×10^{-5}
		I	2×10^{-8}	3×10^{-3}	6×10^{-10}	9×10^{-5}
Americium (95)	Am 241	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		* I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Am 242m	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	3×10^{-10}	3×10^{-3}	9×10^{-12}	9×10^{-5}
	Am 242	S	4×10^{-8}	4×10^{-3}	1×10^{-9}	1×10^{-4}
		I	5×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Am 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Am 244	S	4×10^{-6}	1×10^{-1}	1×10^{-7}	5×10^{-3}
		I	2×10^{-5}	1×10^{-1}	8×10^{-7}	5×10^{-3}
Antimony (51)	Sb 122	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
	Sb 124	S	2×10^{-7}	7×10^{-4}	5×10^{-9}	2×10^{-5}
		I	2×10^{-8}	7×10^{-4}	7×10^{-10}	2×10^{-5}
	Sb 125	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	3×10^{-8}	3×10^{-3}	9×10^{-10}	1×10^{-4}
Argon (18)	A 37	Sub[2]	6×10^{-3}	----------	1×10^{-4}	----------
	A 41	Sub	2×10^{-6}	----------	4×10^{-8}	----------
Arsenic (33)	As 73	S	2×10^{-6}	1×10^{-2}	7×10^{-8}	5×10^{-4}
		I	4×10^{-7}	1×10^{-2}	1×10^{-8}	5×10^{-4}
	As 74	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	5×10^{-5}
		I	1×10^{-7}	2×10^{-3}	4×10^{-9}	5×10^{-5}
	As 76	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	6×10^{-4}	3×10^{-9}	2×10^{-5}
	As 77	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Astatine (85)	At 211	S	7×10^{-9}	5×10^{-5}	2×10^{-10}	2×10^{-6}
		I	3×10^{-8}	2×10^{-5}	1×10^{-9}	7×10^{-5}
Barium (56)	Ba 131	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-8}	2×10^{-4}
	Ba 140	S	1×10^{-7}	8×10^{-4}	4×10^{-9}	3×10^{-5}
		I	4×10^{-8}	7×10^{-4}	1×10^{-9}	2×10^{-5}
Berkelium (97)	Bk 249	S	9×10^{-10}	2×10^{-2}	3×10^{-11}	6×10^{-4}
		I	1×10^{-7}	2×10^{-2}	4×10^{-9}	6×10^{-4}
	Bk 250	S	1×10^{-7}	6×10^{-3}	5×10^{-9}	2×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
Beryllium (4)	Be 7	S	6×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
		I	1×10^{-6}	5×10^{-2}	4×10^{-8}	2×10^{-3}
Bismuth (83)	Bi 206	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		I	1×10^{-7}	1×10^{-3}	5×10^{-9}	4×10^{-5}
	Bi 207	S	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
		I	1×10^{-8}	2×10^{-3}	5×10^{-10}	6×10^{-5}
	Bi 210	S	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
		I	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
	Bi 212	S	1×10^{-7}	1×10^{-2}	3×10^{-9}	4×10^{-4}
		I	2×10^{-7}	1×10^{-2}	7×10^{-9}	4×10^{-4}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Bromine (35)	Br 82	S	1×10^{-6}	8×10^{-3}	4×10^{-8}	3×10^{-4}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Cadmium (48)	Cd 109	S	5×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}
		I	7×10^{-8}	5×10^{-3}	3×10^{-9}	2×10^{-4}
	Cd 115m	S	4×10^{-8}	7×10^{-4}	1×10^{-9}	3×10^{-5}
		I	4×10^{-8}	7×10^{-4}	1×10^{-9}	3×10^{-5}
	Cd 115	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Calcium (20)	Ca 45	S	3×10^{-8}	3×10^{-4}	1×10^{-9}	9×10^{-6}
		I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
	Ca 47	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	3×10^{-5}
Californium (98)	Cf 249	S	2×10^{-12}	1×10^{-4}	5×10^{-14}	4×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cf 250	S	5×10^{-12}	4×10^{-4}	2×10^{-13}	1×10^{-5}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 251	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 252	* S	6×10^{-12}	2×10^{-4}	2×10^{-13}	7×10^{-6}
		* I	3×10^{-11}	2×10^{-4}	1×10^{-12}	7×10^{-6}
	Cf 253	S	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
		I	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
	Cf 254	S	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
		I	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
Carbon (6)	C 14	S	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
	(CO_2)	Sub	5×10^{-5}	----------	1×10^{-6}	----------
Cerium (58)	Ce 141	S	4×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	2×10^{-7}	3×10^{-3}	5×10^{-9}	9×10^{-5}
	Ce 143	S	3×10^{-7}	1×10^{-3}	9×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
	Ce 144	S	1×10^{-8}	3×10^{-4}	3×10^{-10}	1×10^{-5}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
Cesium (55)	Cs 131	S	1×10^{-5}	7×10^{-2}	4×10^{-7}	2×10^{-3}
		I	3×10^{-6}	3×10^{-2}	1×10^{-7}	9×10^{-4}
	Cs 134m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	6×10^{-3}
		I	6×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
	Cs 134	S	4×10^{-8}	3×10^{-4}	1×10^{-9}	9×10^{-6}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	4×10^{-5}
	Cs 135	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	9×10^{-8}	7×10^{-3}	3×10^{-9}	2×10^{-4}
	Cs 136	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	9×10^{-5}
		I	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
	Cs 137	S	6×10^{-8}	4×10^{-4}	2×10^{-9}	2×10^{-5}
		I	1×10^{-8}	1×10^{-3}	5×10^{-10}	4×10^{-5}
Chlorine (17)	Cl 36	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
		I	2×10^{-8}	2×10^{-3}	8×10^{-10}	6×10^{-5}
	Cl 38	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	7×10^{-8}	4×10^{-4}
Chromium (24)	Cr 51	S	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		I	2×10^{-6}	5×10^{-2}	8×10^{-8}	2×10^{-3}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Cobalt (27)	Co 57	S	3×10^{-6}	2×10^{-2}	1×10^{-7}	5×10^{-4}
		I	2×10^{-7}	1×10^{-2}	6×10^{-9}	4×10^{-4}
	Co 58m	S	2×10^{-5}	8×10^{-2}	6×10^{-7}	3×10^{-3}
		I	9×10^{-6}	6×10^{-2}	3×10^{-7}	2×10^{-3}
	Co 58	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-8}	3×10^{-3}	2×10^{-9}	9×10^{-5}
	Co 60	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
		I	9×10^{-9}	1×10^{-3}	3×10^{-10}	3×10^{-5}
Copper (29)	Cu 64	S	2×10^{-6}	1×10^{-2}	7×10^{-8}	3×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
Curium (96)	Cm 242	S	1×10^{-10}	7×10^{-4}	4×10^{-12}	2×10^{-5}
		* I	2×10^{-10}	7×10^{-4}	6×10^{-12}	2×10^{-5}
	Cm 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	5×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cm 244	S	9×10^{-12}	2×10^{-4}	3×10^{-13}	7×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
	Cm 245	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 246	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 247	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	6×10^{-4}	4×10^{-12}	2×10^{-5}
	Cm 248	S	6×10^{-13}	1×10^{-5}	2×10^{-14}	4×10^{-7}
		I	1×10^{-11}	4×10^{-5}	4×10^{-13}	1×10^{-6}
	Cm 249	S	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}
		I	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}
Dysprosium (66)	Dy 165	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	7×10^{-8}	4×10^{-4}
	Dy 166	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
Einsteinium (99)	Es 253	S	8×10^{-10}	7×10^{-4}	3×10^{-11}	2×10^{-5}
		I	6×10^{-10}	7×10^{-4}	2×10^{-11}	2×10^{-5}
	Es 254m	S	5×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
		I	6×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
	Es 254	S	2×10^{-11}	4×10^{-4}	6×10^{-13}	1×10^{-5}
		I	1×10^{-10}	4×10^{-4}	4×10^{-12}	1×10^{-5}
	Es 255	S	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
		I	4×10^{-10}	8×10^{-4}	1×10^{-11}	3×10^{-5}
Erbium (68)	Er 169	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	4×10^{-7}	3×10^{-3}	1×10^{-8}	9×10^{-5}
	Er 171	S	7×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Europium (63)	Eu 152	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	(T/2 = 9.2 hrs)	I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	Eu 152	S	1×10^{-8}	2×10^{-3}	4×10^{-10}	8×10^{-5}
	(T/2 = 13 yrs)	I	2×10^{-8}	2×10^{-3}	6×10^{-10}	8×10^{-5}
	Eu 154	S	4×10^{-9}	6×10^{-4}	1×10^{-10}	2×10^{-5}
		I	7×10^{-9}	6×10^{-4}	2×10^{-10}	2×10^{-5}
	Eu 155	S	9×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
		I	7×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Fermium (100)	Fm 254	S	6×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
		I	7×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Fm 255	S	2×10^{-8}	1×10^{-3}	6×10^{-10}	3×10^{-5}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	3×10^{-5}
	Fm 256	S	3×10^{-9}	3×10^{-3}	1×10^{-10}	9×10^{-7}
		I	2×10^{-9}	3×10^{-3}	6×10^{-11}	9×10^{-7}
Fluorine (9)	F 18	S	5×10^{-6}	2×10^{-2}	2×10^{-7}	8×10^{-4}
		I	3×10^{-6}	1×10^{-2}	9×10^{-8}	5×10^{-4}
Gadolinium (64)	Gd 153	S	2×10^{-7}	6×10^{-3}	8×10^{-9}	2×10^{-4}
		I	9×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Gd 159	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Gallium (31)	Ga 72	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Germanium (32)	Ge 71	S	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		I	6×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
Gold (79)	Au 196	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	6×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
	Au 198	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
	Au 199	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	8×10^{-7}	4×10^{-3}	3×10^{-8}	2×10^{-4}
Hafnium (72)	Hf 181	S	4×10^{-8}	2×10^{-3}	1×10^{-9}	7×10^{-5}
		I	7×10^{-8}	2×10^{-3}	3×10^{-9}	7×10^{-5}
Holmium (67)	Ho 166	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
Hydrogen (1)	H3	S	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
		I	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
		Sub	2×10^{-3}	----------	4×10^{-5}	----------
Indium (49)	In 113m	S	8×10^{-6}	4×10^{-2}	3×10^{-7}	1×10^{-3}
		I	7×10^{-6}	4×10^{-2}	2×10^{-7}	1×10^{-3}
	In 114m	S	1×10^{-7}	5×10^{-4}	4×10^{-9}	2×10^{-5}
		I	2×10^{-8}	5×10^{-4}	7×10^{-10}	2×10^{-5}
	In 115m	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	6×10^{-8}	4×10^{-4}
	In 115	S	2×10^{-7}	3×10^{-3}	9×10^{-9}	9×10^{-5}
		I	3×10^{-8}	3×10^{-3}	1×10^{-9}	9×10^{-5}
Iodine (53)	I 125	S	5×10^{-9}	4×10^{-5}	8×10^{-11}	2×10^{-7}
		I	2×10^{-7}	6×10^{-3}	6×10^{-9}	2×10^{-4}
	I 126	S	8×10^{-9}	5×10^{-5}	9×10^{-11}	3×10^{-7}
		I	3×10^{-7}	3×10^{-3}	1×10^{-8}	9×10^{-5}
	I 129	S	2×10^{-9}	1×10^{-5}	2×10^{-11}	6×10^{-8}
		I	7×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
	I 131	S	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	I 132	S	2×10^{-7}	2×10^{-3}	3×10^{-9}	8×10^{-6}
		I	9×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	I 133	S	3×10^{-8}	2×10^{-4}	4×10^{-10}	1×10^{-6}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
	I 134	S	5×10^{-7}	4×10^{-3}	6×10^{-9}	2×10^{-5}

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Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Iodine (53)	I 134	I	3×10^{-6}	2×10^{-2}	1×10^{-7}	6×10^{-4}
	I 135	S	1×10^{-7}	7×10^{-4}	1×10^{-9}	4×10^{-6}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
Iridium (77)	Ir 190	S	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-8}	2×10^{-4}
	Ir 192	S	1×10^{-7}	1×10^{-3}	4×10^{-9}	4×10^{-5}
		I	3×10^{-8}	1×10^{-3}	9×10^{-10}	4×10^{-5}
	Ir 194	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
		I	2×10^{-7}	9×10^{-4}	5×10^{-9}	3×10^{-5}
Iron (26)	Fe 55	S	9×10^{-7}	2×10^{-2}	3×10^{-8}	8×10^{-4}
		I	1×10^{-6}	7×10^{-2}	3×10^{-8}	2×10^{-3}
	Fe 59	S	1×10^{-7}	2×10^{-3}	5×10^{-9}	6×10^{-5}
		I	5×10^{-8}	2×10^{-3}	2×10^{-9}	5×10^{-5}
Krypton (36)	Kr 85m	Sub	6×10^{-6}	----------	1×10^{-7}	----------
	Kr 85	Sub	1×10^{-5}	----------	3×10^{-7}	----------
	Kr 87	Sub	1×10^{-6}	----------	2×10^{-8}	----------
	Kr 88	Sub	1×10^{-6}	----------	2×10^{-8}	----------
Lanthanum (57)	La 140	S	2×10^{-7}	7×10^{-4}	5×10^{-9}	2×10^{-5}
		I	1×10^{-7}	7×10^{-4}	4×10^{-9}	2×10^{-5}
Lead (82)	Pb 203	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	6×10^{-8}	4×10^{-4}
	Pb 210	S	1×10^{-10}	4×10^{-6}	4×10^{-12}	1×10^{-7}
		I	2×10^{-10}	5×10^{-6}	8×10^{-12}	2×10^{-4}
	Pb 212	S	2×10^{-8}	6×10^{-4}	6×10^{-10}	2×10^{-5}
		I	2×10^{-8}	5×10^{-4}	7×10^{-10}	2×10^{-5}
Lutetium (71)	Lu 177	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Manganese (25)	Mn 52	S	2×10^{-7}	1×10^{-3}	7×10^{-9}	3×10^{-5}
		I	1×10^{-7}	9×10^{-4}	5×10^{-9}	3×10^{-5}
	Mn 54	* S	4×10^{-7}	4×10^{-3}	1×10^{-8}	1×10^{-4}
		I	4×10^{-8}	3×10^{-3}	1×10^{-9}	1×10^{-4}
	Mn 56	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Mercury (80)	Hg 197m	S	7×10^{-7}	6×10^{-3}	3×10^{-8}	2×10^{-4}
		I	8×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	Hg 197	S	1×10^{-6}	9×10^{-3}	4×10^{-8}	3×10^{-4}
		I	3×10^{-6}	1×10^{-2}	9×10^{-8}	5×10^{-4}
	Hg 203	S	7×10^{-8}	5×10^{-4}	2×10^{-9}	2×10^{-5}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
Molybdenum (42)	Mo 99	S	7×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
Neodymium (60)	Nd 144	S	8×10^{-11}	2×10^{-3}	3×10^{-12}	7×10^{-5}
		I	3×10^{-10}	2×10^{-3}	1×10^{-11}	8×10^{-5}
	Nd 147	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	2×10^{-7}	2×10^{-3}	8×10^{-9}	6×10^{-5}
	Nd 149	S	2×10^{-6}	8×10^{-3}	6×10^{-8}	3×10^{-4}
		I	1×10^{-6}	8×10^{-3}	5×10^{-8}	3×10^{-4}

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Neptunium (93)	Np 237	S	4×10^{-12}	9×10^{-5}	1×10^{-13}	3×10^{-6}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
	Np 239	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	7×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
Nickel (28)	Ni 59	S	5×10^{-7}	6×10^{-3}	2×10^{-8}	2×10^{-4}
		I	8×10^{-7}	6×10^{-2}	3×10^{-8}	2×10^{-3}
	Ni 63	S	6×10^{-8}	8×10^{-4}	2×10^{-9}	3×10^{-5}
		I	3×10^{-7}	2×10^{-2}	1×10^{-8}	7×10^{-4}
	Ni 65	S	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Niobium (Columbium) (41)	Nb 93m	S	1×10^{-7}	1×10^{-2}	4×10^{-9}	4×10^{-4}
		I	2×10^{-7}	1×10^{-2}	5×10^{-9}	4×10^{-4}
	Nb 95	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	1×10^{-7}	3×10^{-3}	3×10^{-9}	1×10^{-4}
	Nb 97	S	6×10^{-6}	3×10^{-2}	2×10^{-7}	9×10^{-4}
		I	5×10^{-6}	3×10^{-2}	2×10^{-7}	9×10^{-4}
Osmium (76)	Os 185	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	5×10^{-8}	2×10^{-3}	2×10^{-9}	7×10^{-5}
	Os 191m	S	2×10^{-5}	7×10^{-2}	6×10^{-7}	3×10^{-3}
		I	9×10^{-6}	7×10^{-2}	3×10^{-7}	2×10^{-3}
	Os 191	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-8}	2×10^{-4}
	Os 193	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	3×10^{-7}	2×10^{-3}	9×10^{-9}	5×10^{-5}
Palladium (46)	Pd 103	S	1×10^{-6}	1×10^{-2}	5×10^{-8}	3×10^{-4}
		I	7×10^{-7}	8×10^{-3}	3×10^{-8}	3×10^{-4}
	Pd 109	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
Phosphorus (15)	P 32	S	7×10^{-8}	5×10^{-4}	2×10^{-9}	2×10^{-5}
		I	8×10^{-8}	7×10^{-4}	3×10^{-9}	2×10^{-5}
Platinum (78)	Pt 191	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
	Pt 193m	S	7×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
		I	5×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
*	Pt 193	S	1×10^{-6}	3×10^{-2}	4×10^{-8}	9×10^{-4}
		I	3×10^{-7}	5×10^{-2}	1×10^{-8}	2×10^{-3}
	Pt 197m	S	6×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
		I	5×10^{-6}	3×10^{-2}	2×10^{-7}	9×10^{-4}
	Pt 197	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Plutonium (94)	Pu 238	S	2×10^{-12}	1×10^{-4}	7×10^{-14}	5×10^{-6}
		I	3×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-5}
	Pu 239	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-5}
	Pu 240	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-5}
	Pu 241	S	9×10^{-11}	7×10^{-3}	3×10^{-12}	2×10^{-4}
		I	4×10^{-8}	4×10^{-2}	1×10^{-9}	1×10^{-3}

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Plutonium (94)	Pu 242	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	9×10^{-4}	1×10^{-12}	3×10^{-5}
	Pu 243	S	2×10^{-6}	1×10^{-2}	6×10^{-8}	3×10^{-4}
		I	2×10^{-6}	1×10^{-2}	8×10^{-8}	3×10^{-4}
	Pu 244	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	3×10^{-11}	3×10^{-4}	1×10^{-12}	1×10^{-5}
Polonium (84)	Po 210	S	5×10^{-10}	2×10^{-5}	2×10^{-11}	7×10^{-7}
		I	2×10^{-10}	8×10^{-4}	7×10^{-12}	3×10^{-5}
Potassium (19)	K 42	S	2×10^{-6}	9×10^{-3}	7×10^{-8}	3×10^{-4}
		I	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
Praseodymium (59)	Pr 142	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	2×10^{-7}	9×10^{-4}	5×10^{-9}	3×10^{-5}
	Pr 143	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	5×10^{-5}
Promethium (61)	Pm 147	S	6×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
		I	1×10^{-7}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Pm 149	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
Protoactinium (91)	Pa 230	S	2×10^{-9}	7×10^{-3}	6×10^{-11}	2×10^{-4}
		I	8×10^{-10}	7×10^{-3}	3×10^{-11}	2×10^{-4}
	Pa 231	S	1×10^{-12}	3×10^{-5}	4×10^{-14}	9×10^{-7}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	2×10^{-5}
	Pa 233	S	6×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
		I	2×10^{-7}	3×10^{-3}	6×10^{-9}	1×10^{-4}
Radium (88)	Ra 223	S	2×10^{-9}	2×10^{-5}	6×10^{-11}	7×10^{-7}
		I	2×10^{-10}	1×10^{-4}	8×10^{-12}	4×10^{-6}
	Ra 224	S	5×10^{-9}	7×10^{-5}	2×10^{-10}	2×10^{-6}
		I	7×10^{-10}	2×10^{-4}	2×10^{-11}	5×10^{-6}
	Ra 226	S	3×10^{-11}	4×10^{-7}	3×10^{-12}	3×10^{-8}
		I	5×10^{-11}	9×10^{-4}	2×10^{-12}	3×10^{-5}
	Ra 228	S	7×10^{-11}	8×10^{-7}	2×10^{-12}	3×10^{-8}
		I	4×10^{-11}	7×10^{-4}	1×10^{-12}	3×10^{-5}
Radon (86)	Rn 220	S	3×10^{-7}	----------	1×10^{-8}	----------
	Rn 222	S	1×10^{-7}	----------	3×10^{-9}	----------
Rhenium (75)	Re 183	S	3×10^{-6}	2×10^{-2}	9×10^{-8}	6×10^{-4}
		I	2×10^{-7}	8×10^{-3}	5×10^{-9}	3×10^{-4}
	Re 186	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
	Re 187	S	9×10^{-6}	7×10^{-2}	3×10^{-7}	3×10^{-3}
		I	5×10^{-7}	4×10^{-2}	2×10^{-8}	2×10^{-3}
	Re 188	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
Rhodium (45)	Rh 103m	S	8×10^{-5}	4×10^{-1}	3×10^{-6}	1×10^{-2}
		I	6×10^{-5}	3×10^{-1}	2×10^{-6}	1×10^{-2}
	Rh 105	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Rubidium (37)	Rb 86	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
		I	7×10^{-8}	7×10^{-4}	2×10^{-9}	2×10^{-5}
	Rb 87	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	7×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}

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Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Ruthenium (44)	Ru 97	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	6×10^{-8}	3×10^{-4}
	Ru 103	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	8×10^{-8}	2×10^{-3}	3×10^{-9}	8×10^{-5}
	Ru 105	S	7×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
	Ru 106	S	8×10^{-8}	4×10^{-4}	3×10^{-9}	1×10^{-5}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
Samarium (62)	Sm 147	S	7×10^{-11}	2×10^{-3}	2×10^{-12}	6×10^{-5}
		I	3×10^{-10}	2×10^{-3}	9×10^{-12}	7×10^{-5}
	Sm 151	S	6×10^{-8}	1×10^{-2}	2×10^{-9}	4×10^{-4}
		I	1×10^{-7}	1×10^{-2}	5×10^{-9}	4×10^{-4}
	Sm 153	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Scandium (21)	Sc 46	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-8}	1×10^{-3}	8×10^{-10}	4×10^{-5}
	Sc 47	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
	Sc 48	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
Selenium (34)	Se 75	S	1×10^{-6}	9×10^{-3}	4×10^{-8}	3×10^{-4}
		I	1×10^{-7}	8×10^{-3}	4×10^{-9}	3×10^{-4}
Silicon (14)	Si 31	S	6×10^{-6}	3×10^{-2}	2×10^{-7}	9×10^{-4}
		I	1×10^{-6}	6×10^{-3}	3×10^{-8}	2×10^{-4}
Silver (47)	Ag 105	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	8×10^{-8}	3×10^{-3}	3×10^{-9}	1×10^{-4}
	Ag 110m	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	1×10^{-8}	9×10^{-4}	3×10^{-10}	3×10^{-5}
	Ag 111	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
Sodium (11)	Na 22	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		I	9×10^{-9}	9×10^{-4}	3×10^{-10}	3×10^{-5}
	Na 24	S	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
Strontium (38)	Sr 85m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
		I	3×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
	Sr 85	S	2×10^{-7}	3×10^{-3}	8×10^{-9}	1×10^{-4}
		I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
	Sr 89	S	3×10^{-8}	3×10^{-4}	3×10^{-10}	3×10^{-6}
		I	4×10^{-8}	8×10^{-4}	1×10^{-9}	3×10^{-5}
	Sr 90	S	1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-7}
		I	5×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
	Sr 91	S	4×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	3×10^{-7}	1×10^{-3}	9×10^{-9}	5×10^{-5}
	Sr 92	S	4×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
Sulfur (16)	S 35	S	3×10^{-7}	2×10^{-3}	9×10^{-9}	6×10^{-5}
		I	3×10^{-7}	8×10^{-3}	9×10^{-9}	3×10^{-4}
Tantalum (73)	Ta 182	S	4×10^{-8}	1×10^{-3}	1×10^{-9}	4×10^{-5}
		I	2×10^{-8}	1×10^{-3}	7×10^{-10}	4×10^{-5}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Technetium (43)	Tc 96m	S	8×10^{-5}	4×10^{-1}	3×10^{-6}	1×10^{-2}
		I	3×10^{-5}	3×10^{-1}	1×10^{-6}	1×10^{-2}
	Tc 96	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
	Tc 97m	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	2×10^{-7}	5×10^{-3}	5×10^{-9}	2×10^{-4}
	Tc 97	S	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		I	3×10^{-7}	2×10^{-2}	1×10^{-8}	8×10^{-4}
	Tc 99m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	6×10^{-3}
		I	1×10^{-5}	8×10^{-2}	5×10^{-7}	3×10^{-3}
	Tc 99	S	2×10^{-6}	1×10^{-2}	7×10^{-8}	3×10^{-4}
		I	6×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}
Tellurium (52)	Te 125m	S	4×10^{-7}	5×10^{-3}	1×10^{-8}	2×10^{-4}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
	Te 127m	S	1×10^{-7}	2×10^{-3}	5×10^{-9}	6×10^{-5}
		I	4×10^{-8}	2×10^{-3}	1×10^{-9}	5×10^{-5}
	Te 127	S	2×10^{-6}	8×10^{-3}	6×10^{-8}	3×10^{-4}
		I	9×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	Te 129m	S	8×10^{-8}	1×10^{-3}	3×10^{-9}	3×10^{-5}
		I	3×10^{-8}	6×10^{-4}	1×10^{-9}	2×10^{-5}
	Te 129	S	5×10^{-6}	2×10^{-2}	2×10^{-7}	8×10^{-4}
		I	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
	Te 131m	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
	Te 132	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
Terbium (65)	Tb 160	S	1×10^{-7}	1×10^{-3}	3×10^{-9}	4×10^{-5}
		I	3×10^{-8}	1×10^{-3}	1×10^{-9}	4×10^{-5}
Thallium (81)	Tl 200	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	1×10^{-6}	7×10^{-3}	4×10^{-8}	2×10^{-4}
	Tl 201	S	2×10^{-6}	9×10^{-3}	7×10^{-8}	3×10^{-4}
		I	9×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	Tl 202	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	2×10^{-7}	2×10^{-3}	8×10^{-9}	7×10^{-5}
	Tl 204	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	3×10^{-8}	2×10^{-3}	9×10^{-10}	6×10^{-5}
Thorium (90) *	Th 227	S	3×10^{-10}	5×10^{-4}	1×10^{-11}	2×10^{-5}
		I	2×10^{-10}	5×10^{-4}	6×10^{-12}	2×10^{-5}
	Th 228	S	9×10^{-12}	2×10^{-4}	3×10^{-13}	7×10^{-6}
		I	6×10^{-12}	4×10^{-4}	2×10^{-13}	1×10^{-5}
	Th 230	S	2×10^{-12}	5×10^{-5}	8×10^{-14}	2×10^{-6}
		I	1×10^{-11}	9×10^{-4}	3×10^{-13}	3×10^{-6}
*	Th 231	S	1×10^{-6}	7×10^{-3}	5×10^{-8}	2×10^{-4}
		I	1×10^{-6}	7×10^{-3}	4×10^{-8}	2×10^{-4}
	Th 232	S	3×10^{-11}	5×10^{-5}	1×10^{-12}	2×10^{-6}
		I	3×10^{-11}	1×10^{-3}	1×10^{-12}	4×10^{-5}
**	Th natural	S	6×10^{-11}	6×10^{-5}	2×10^{-12}	2×10^{-6}
		I	6×10^{-11}	6×10^{-4}	2×10^{-12}	2×10^{-5}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]	†	Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Thorium (90)	Th 234	S	6×10^{-8}	5×10^{-4}	2×10^{-9}	2×10^{-5}
		I	3×10^{-8}	5×10^{-4}	1×10^{-9}	2×10^{-5}
Thulium (69)	Tm 170	S	4×10^{-8}	1×10^{-3}	1×10^{-9}	5×10^{-5}
		I	3×10^{-8}	1×10^{-3}	1×10^{-9}	5×10^{-5}
	Tm 171	S	1×10^{-7}	1×10^{-2}	4×10^{-9}	5×10^{-4}
		I	2×10^{-7}	1×10^{-2}	8×10^{-9}	5×10^{-4}
Tin (50)	Sn 113	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	9×10^{-5}
		I	5×10^{-8}	2×10^{-3}	2×10^{-9}	8×10^{-5}
	Sn 125	S	1×10^{-7}	5×10^{-4}	4×10^{-9}	2×10^{-5}
		I	8×10^{-8}	5×10^{-4}	3×10^{-9}	2×10^{-5}
Tungsten (Wolfram) (74)	W 181	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	1×10^{-7}	1×10^{-2}	4×10^{-9}	3×10^{-4}
	W 185	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
	W 187	S	4×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
Uranium (92)	U 230	S	3×10^{-10}	1×10^{-4}	1×10^{-11}	5×10^{-6}
		I	1×10^{-10}	1×10^{-4}	4×10^{-12}	5×10^{-6}
	U 232	S	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
		I	3×10^{-11}	8×10^{-4}	9×10^{-13}	3×10^{-5}
	U 233	S	5×10^{-10}	9×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
**	U 234	S[3]	6×10^{-10}	9×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
**	U 235	S[3]	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	U 236	S	6×10^{-10}	1×10^{-3}	2×10^{-11}	4×10^{-5}
		I	1×10^{-10}	1×10^{-3}	4×10^{-12}	3×10^{-5}
**	U 238	S[3]	7×10^{-11}	1×10^{-3}	3×10^{-12}	4×10^{-5}
		I	1×10^{-10}	1×10^{-3}	5×10^{-12}	4×10^{-5}
	U 240	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	3×10^{-5}
**	U-natural	S[3]	1×10^{-10}	1×10^{-3}	5×10^{-12}	3×10^{-5}
		I	1×10^{-10}	1×10^{-3}	5×10^{-12}	3×10^{-5}
Vanadium (23)	V 48	S	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
		I	6×10^{-8}	8×10^{-4}	2×10^{-9}	3×10^{-5}
Xenon (54)	Xe 131m	Sub	2×10^{-5}	----------	4×10^{-7}	----------
	Xe 133	Sub	1×10^{-5}	----------	3×10^{-7}	----------
	Xe 133m	Sub	1×10^{-5}	----------	3×10^{-7}	----------
	Xe 135	Sub	4×10^{-6}	----------	1×10^{-7}	----------
Ytterbium (70)	Yb 175	S	7×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Yttrium (39)	Y 90	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	6×10^{-4}	3×10^{-9}	2×10^{-5}
	Y 91m	S	2×10^{-5}	1×10^{-1}	8×10^{-7}	3×10^{-3}
		I	2×10^{-5}	1×10^{-1}	6×10^{-7}	3×10^{-3}
	Y 91	S	4×10^{-8}	8×10^{-4}	1×10^{-9}	3×10^{-5}
		I	3×10^{-8}	8×10^{-4}	1×10^{-9}	3×10^{-5}
	Y 92	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	Y 93	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See footnotes on page 20-15)

Element (atomic number)	Isotope [1]		Table I Column 1 Air (μCi/ml)	Table I Column 2 Water (μCi/ml)	Table II Column 1 Air (μCi/ml)	Table II Column 2 Water (μCi/ml)
Zinc (30)	Zn 65	S	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
		I	6×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}
	Zn 69m	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	Zn 69	S	7×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
		I	9×10^{-6}	5×10^{-2}	3×10^{-7}	2×10^{-3}
Zirconium (40)	Zr 93	S	1×10^{-7}	2×10^{-2}	4×10^{-9}	8×10^{-4}
		I	3×10^{-7}	2×10^{-2}	1×10^{-8}	8×10^{-4}
	Zr 95	S	1×10^{-7}	2×10^{-3}	4×10^{-9}	6×10^{-5}
		I	3×10^{-8}	2×10^{-3}	1×10^{-9}	6×10^{-5}
	Zr 97	S	1×10^{-7}	5×10^{-4}	4×10^{-9}	2×10^{-5}
		I	9×10^{-8}	5×10^{-4}	3×10^{-9}	2×10^{-5}
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life less than 2 hours.		Sub	1×10^{-6}		3×10^{-8}	
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life greater than 2 hours.			3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}
Any single radionuclide not listed above, which decays by alpha emission or spontaneous fission.			6×10^{-13}	4×10^{-7}	2×10^{-14}	3×10^{-8}

[1] Soluble (S); Insoluble (I).

[2] "Sub" means that values given are for submersion in a semispherical infinite cloud of airborne material.

‡3. For soluble mixtures of U-238, U-234 and U-235 in air chemical toxicity may be the limiting factor. If the percent by weight (enrichment) of U-235 is less than 5, the concentration value for a 40-hour workweek, Table I, is 0.2 milligrams uranium per cubic meter of air average. For any enrichment, the product of the average concentration and time of exposure during a 40-hour workweek shall not exceed 8×10^{-3} SA μCi-hr/ml, where SA is the specific activity of the uranium inhaled. The concentration value for Table II is 0.007 milligrams uranium per cubic meter of air. The specific activity for natural uranium is 6.77×10^{-7} curies per gram U. The specific activity for other mixtures of U-238, U-235 and U-234, if not known, shall be:

$SA = 3.6\times10^{-7}$ curies/gram U U-depleted

$SA = (0.4 + 0.38\,E + 0.0034\,E^2)\,10^{-6}$ $E \geq 0.72$

where E is the percentage by weight of U-235, expressed as percent.

* Amended 37 FR 23319.
** Amended 39 FR 23990.
† Amended 38 FR 29314.
‡ Amended 39 FR 25463.

25 FR 10914

39 FR 23990

NOTE TO APPENDIX B

25 FR 10914

NOTE: In any case where there is a mixture in air or water of more than one radionuclide, the limiting values for purposes of this Appendix should be determined as follows:

1. If the identity and concentration of each radionuclide in the mixture are known, the limiting values should be derived as follows: Determine, for each radionuclide in the mixture, the ratio between the quantity present in the mixture and the limit otherwise established in Appendix B for the specific radionuclide when not in a mixture. The sum of such ratios for all the radionuclides in the mixture may not exceed "1" (i.e., "unity").

EXAMPLE: If radionuclides A, B, and C are present in concentrations C_A, C_B, and C_C, and if the applicable MPC's, are MPC_A, and MPC_B, and MPC_C respectively, then the concentrations shall be limited so that the following relationship exists:

$$\frac{C_A}{MPC_A}+\frac{C_B}{MPC_B}+\frac{C_C}{MPC_C}\leq 1$$

30 FR 15801

2. If either the identity or the concentration of any radionuclide in the mixture is not known, the limiting values for purposes of Appendix B shall be:

a. For purposes of Table I, Col. 1—6×10^{-13}

b. For purposes of Table I, Col. 2—4×10^{-7}

c. For purposes of Table II, Col. 1—2×10^{-14}

d. For purposes of Table II, Col. 2—3×10^{-8}

26 FR 11046

3. If any of the conditions specified below are met, the corresponding values specified below may be used in lieu of those specified in paragraph 2 above.

a. If the identity of each radionuclide in the mixture is known but the concentration of one or more of the radionuclides in the mixture is not known, the concentration limit for the mixture is the limit specified in Appendix "B" for the radionuclide in the mixture having the lowest concentration limit; or

b. If the identity of each radionuclide in the mixture is not known, but it is known that certain radionuclides specified in Appendix "B" are not present in the mixture, the concentration limit for the mixture is the lowest concentration limit specified in Appendix "B" for any radionuclide which is not known to be absent from the mixture; or

30 FR 15801

c. Element (atomic number) and isotope	Table I		Table II	
	Column 1 Air (μCi/ml)	Column 2 Water (μCi/ml)	Column 1 Air (μCi/ml)	Column 2 Water (μCi/ml)
If it is known that Sr 90, I 125, I 126, I 129, I 131, (I 133, table II only), Pb 210, Po 210, At 211, Ra 223, Ra 224, Ra 226, Ac 227, Ra 228, Th 230, Pa 231, Th 232, Th-nat, Cm 248, Cf 254, and Fm 256 are not present		9×10^{-8}		3×10^{-6}
If it is known that Sr 90, I 125, I 126, I 129, (I 131, I 133, table II only), Pb 210, Po 210, Ra 223, Ra 226, Ra 228, Pa 231, Th-nat, Cm 248, Cf 254, and Fm 256 are not present		6×10^{-8}		2×10^{-6}
If it is known that Sr 90, I 129, (I 125, I 126, I 131, table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present		2×10^{-8}		6×10^{-7}
If it is known that (I 129, table II only), Ra 226, and Ra 228 are not present		3×10^{-8}		1×10^{-7}
If it is known that alpha-emitters and Sr 90, I 129, Pb 210, Ac 227, Ra 228, Pa 230, Pu 241, and Bk 249 are not present	3×10^{-9}		1×10^{-10}	
If it is known that alpha-emitters and Pb 210, Ac 227, Ra 228, and Pu 241 are not present	3×10^{-10}		1×10^{-11}	
If it is known that alpha-emitters and Ac 227 are not present	3×10^{-11}		1×10^{-12}	
If it is known that Ac 227, Th 230, Pa 231, Pu 238, Pu 239, Pu 240, Pu 242, Pu 244, Cm 248, Cf 249 and Cf 251 are not present	3×10^{-12}		1×10^{-12}	

25 FR 13952

4. If the mixture of radionuclides consists of uranium and its daughter products in ore dust prior to chemical processing of the uranium ore, the values specified below may be used in lieu of those determined in accordance with paragraph 1 above or those specified in paragraphs 2 and 3 above.

39 FR 23990

a. For purposes of Table I, Col. 1—1×10^{-10} μCi/ml gross alpha activity; or 5×10^{-11} μCi/ml natural uranium; or 75 micrograms per cubic meter of air natural uranium.

b. For purposes of Table II, Col. 1—3×10^{-12} μCi/ml gross alpha activity; or 2×10^{-12} μCi/ml natural uranium; or 3 micrograms per cubic meter of air natural uranium.

26 FR 11046

5. For purposes of this note, a radionuclide may be considered as not present in a mixture if (a) the ratio of the concentration of that radionuclide in the mixture (C_A) to the concentration limit for that radionuclide specified in Table II of Appendix B (MPC_A) does not exceed 1/10 (i.e. $\frac{C_A}{MPC_A}\leq\frac{1}{10}$) and (b) the sum of such ratios for all the radionuclides considered as not present in the mixture does not exceed 1/4 (i.e. $\frac{C_A}{MPC_A}+\frac{C_B}{MPC_B}+\ldots\ldots\leq\frac{1}{4}$).

35 FR 6425

APPENDIX C

Material	Microcuries
Americium-241	.01
Antimony-122	100
Antimony-124	10
Antimony-125	10
Arsenic-73	100
Arsenic-74	10
Arsenic-76	10
Arsenic-77	100
Barium-131	10
*Barium-133	10
Barium-140	10
Bismuth-210	1
Bromine-82	10
Cadmium-109	10
Cadmium-115m	10
Cadmium-115	100
Calcium-45	10
Calcium-47	10
Carbon-14	100
Cerium-141	100
Cerium-143	100
Cerium-144	1
Cesium-131	1,000
Cesium-134m	100
Cesium-134	1
Cesium-135	10
Cesium-136	10
Cesium-137	10
Chlorine-36	10
Chlorine-38	10
Chromium-51	1,000
Cobalt-58m	10
Cobalt-58	10
Cobalt-60	1
Copper-64	100
Dysprosium-165	10
Dysprosium-166	100
Erbium-169	100
Erbium-171	100
Europium-152 9.2 h	100
Europium-152 13 yr	1
Europium-154	1
Europium-155	10
Fluorine-18	1,000
Gadolinium-153	10
Gadolinium-159	100
Gallium-72	10
Germanium-71	100
Gold-198	100
Gold-199	100
Hafnium-181	10
Holmium-166	100
Hydrogen-3	1,000
Indium-113m	100
Indium-114m	10
Indium-115m	100
Indium-115	10
Iodine-125	1
Iodine-126	1
Iodine-129	0.1
Iodine-131	1
Iodine-132	10
Iodine-133	1
Iodine-134	10
Iodine-135	10
Iridium-192	10
Iridium-194	100
Iron-55	100
Iron-59	10
Krypton-85	100
Krypton-87	10
Lanthanum-140	10
Lutetium-177	100
Manganese-52	10
Manganese-54	10
Manganese-56	10
Mercury-197m	100
Mercury-197	100
Mercury-203	10
Molybdenum-99	100
Neodymium-147	100
Neodymium-149	100
Nickel-59	100
Nickel-63	10
Nickel-65	100
Niobium-93m	10
Niobium-95	10
Niobium-97	10
Osmium-185	10

35 FR 6425

Material	Microcuries
Osmium-191m*	100
Osmium-191	100
Osmium-193	100
Palladium-103	100
Palladium-109	100
Phosphorus-32	10
Platinum-191	100
Platinum-193m	100
Platinum-193	100
Platinum-197m	100
Platinum-197	100
Plutonium-239	.01
Polonium-210	0.1
Potassium-42	10
Praseodymium-142	100
Praseodymium-143	100
Promethium-147	10
Promethium-149	10
Radium-226	.01
Rhenium-186	100
Rhenium-188	100
Rhodium-103m	100
Rhodium-105	100
Rubidium-86	10
Rubidium-87	10
Ruthenium-97	100
Ruthenium-103	10
Ruthenium-105	10
Ruthenium-106	1
Samarium-151	10
Samarium-153	100
Scandium-46	10
Scandium-47	100
Scandium-48	10
Selenium-75	10
Silicon-31	100
Silver-105	10
Silver-110m	1
Silver-111	100
Sodium-24	10
Strontium-85	10
Strontium-89	1
Strontium-90	0.1
Strontium-91	10
Strontium-92	10
Sulphur-35	100
Tantalum-182	10
Technetium-96	10
Technetium-97m	100
Technetium-97	100
Technetium-99m	100
Technetium-99	10
Tellurium-125m	10
Tellurium-127m	10
Tellurium-127	100
Tellurium-129m	10
Tellurium-129	100
Tellurium-131m	10
Tellurium-132	10
Terbium-160	10
Thallium-200	100
Thallium-201	100
Thallium-202	100
Thallium-204	10
**Thorium (natural)[1]	100
Thulium-170	10
Thulium-171	10
Tin-113	10
Tin-125	10
Tungsten-181	10
Tungsten-185	10
Tungsten-187	100
**Uranium (natural)[2]	100
Uranium-233	.01
Uranium-234—Uranium-235	.01
Vanadium-48	10
Xenon-131m	1,000
Xenon-133	100
Xenon-135	100
Ytterbium-175	100
Yttrium-90	10
Yttrium-91	10
Yttrium-92	100
Yttrium-93	100
Zinc-65	10
Zinc-69m	100
Zinc-69	1,000
Zirconium-93	10
Zirconium-95	10
Zirconium-97	10

35 FR 6425

Any alpha emitting radionuclide not listed above or mixtures of alpha emitters of unknown composition	.01
Any radionuclide other than alpha emitting radionuclides, not listed above or mixtures of beta emitters of unknown composition	.1

NOTE: For purposes of §§ 20.203 and 20.304, where there is involved a combination of isotopes in known amounts the limit for the combination should be derived as follows: Determine, for each isotope in the combination, the ratio between the quantity present in the combination and the limit otherwise established for the specific isotope when not in combination. The sum of such ratios for all the isotopes in the combination may not exceed "1" (i.e., "unity"). Example: For purposes of § 20.304, if a particular batch contains 20,000 μCi† of Au^{198} and 50,000 μCi† of C^{14}, it may also include not more than 300 μCi† of I^{131}. This limit was determined as follows:

$$\dagger\ \frac{20{,}000\ \mu Ci\ Au^{198}}{100{,}000\ \mu Ci} + \frac{50{,}000\ \mu Ci\ C^{14}}{100{,}000\ \mu Ci} + \frac{300\ \mu Ci\ I^{131}}{1{,}000\ \mu Ci} = 1$$

The denominator in each of the above ratios was obtained by multiplying the figure in the table by 1,000 as provided in § 20.304.

[1] Based on alpha disintegration rate of Th-232, Th-230 and their daughter products.
[2] Based on alpha disintegration rate of U-238, U-234, and U-235.
* Amended 36 FR 16898.
** Amended 39 FR 23990.
† Amended 38 FR 29314.

Appendix D

UNITED STATES NUCLEAR REGULATORY COMMISSION

INSPECTION AND ENFORCEMENT REGIONAL OFFICES

Region	Address	Telephone	
		Daytime	Nights and Holidays
I Connecticut, Delaware, District of Columbia, Maine, Maryland, Massachusetts, New Hampshire, New Jersey, New York, Pennsylvania, Rhode Island, and Vermont	Region I, USNRC Office of Inspection and Enforcement 631 Park Avenue King of Prussia, Pa. 19406	(215)-337-1150	(215) 337-1150
II* Alabama, Florida, Georgia, Kentucky, Mississippi, North Carolina, Panama Canal Zone, Puerto Rico, South Carolina, Tennessee, Virginia, Virgin Islands, and West Virginia	Region II, USNRC Office of Inspection and Enforcement 230 Peachtree St., N.W. Suite 818 Atlanta, Ga. 30303	(404) 526-4503	(404) 526-4503
III Illinois, Indiana, Iowa, Michigan, Minnesota, Missouri, Ohio, and Wisconsin	Region III, USNRC Office of Inspection and Enforcement 799 Roosevelt Road Glen Ellyn, Ill. 60137	(312) 858-2660	*(312) 739-7711
IV* Arkansas, Colorado, Idaho, Kansas, Louisiana, Montana, Nebraska, New Mexico, North Dakota, Oklahoma, South Dakota, Texas, Utah, and Wyoming	* Region IV, USNRC Office of Inspection and Enforcement 611 Ryan Plaza Drive Suite 1000 Arlington, Texas 76012	*(817) 334-2841	*(817) 334-2841
V Alaska, Arizona, California, Hawaii, Nevada, Oregon, Washington, and U.S. territories and possessions in the Pacific	* Region V, USNRC Office of Inspection and Enforcement 1990 N. California Blvd. Suite 202 Walnut Creek, Calif. 94596	**(415) 486-3141	(415) 273-4237

38 FR 17198

* Amended.
** Amended 39 FR 17972.

NOTE: The reporting and record keeping requirements contained in §§20.205(b) and 20.205(c) and required by §20.401(b) have been approved by GAO under B-180225 (R0054). The approval expires June 30, 1977.

Form NRC-396
(2-75)
10 CFR 55

APPENDIX D

Form approved,
Bureau of Budget No. 38-R084.

U. S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF MEDICAL HISTORY

Facility Operator's or Senior Operator's License

Instructions: Applicant must complete all items on page 1. Typewrite or print in ink. Physician must complete all items on page 2.

1. Last Name | First Name | Middle Name | 2. Date of Birth

3. Home Address | 4. Sex

Have you ever had or do you now have any of the following? Give details of any condition answered in the affirmative under item 37.

	Yes	No		Yes	No
5. Rheumatic fever			16. Bone, joint, or other deformity		
6. Frequent or severe headaches			17. Painful or "trick" shoulder		
7. Dizziness or fainting spells			18. Painful or "trick" elbow		
8. Eye trouble			19. Paralysis		
9. Diabetes			20. Epilepsy or fits		
10. Tuberculosis			21. Depression or excessive worry		
11. Chronic shortness of breath			22. Loss of memory or amnesia		
12. Pain or pressure in chest or "heart attack"			23. Nervous condition which could impair judgment or reliability		
13. High blood pressure					
14. Low blood pressure			24. Drug, narcotic habit or excessive drinking		
15. Peptic ulcer			25. Do you normally wear eyeglasses?		

Complete each of the following. Give details of every affirmative answer under item 37.

	Yes	No
26. Has your work ever been limited or restricted for medical reasons?		
27. Have you ever been denied or rated up for life insurance for medical reasons?		
28. Have you ever been under observation or received care or treatment for any mental or nervous condition as a patient in a hospital, sanitorium, clinic, or other facility, or from a physician, clinical psychologist, etc.?		
29. Have you ever been rejected for or discharged from employment or military service for physical, mental, or nervous disorder reasons?		
30. Have you ever received, is there pending, have you applied for, or do you intend to apply for pension or compensation for existing disability?		
31. Have you ever seriously considered committing suicide?		
32. Have you ever been convicted of any violation of Federal law, State law, county or municipal law, regulations or ordinance? Do not include anything that happened nefore your 16th birthday. Do not include violations for which a fine of $25 or less was imposed.		
33. Have you ever had any major illness or injury other than those already noted?		
34. How many jobs have you had in the last 3 years?		
35. What is the length of time in your present employment?		

36. Give a brief statement of your present health in your own words:

37. Details of any items 5 through 33 answered in the affirmative. In addition, if your medical history includes any matter relating to physical, mental, or nervous condition, please describe the condition and set forth your explanation of why this matter would not affect your ability to function as a facility operator. Use additional sheet if more space is needed.

38. I certify that the foregoing information supplied by me is true to the best of my knowledge, and authorize the Nuclear Regulatory Commission to use any of the information in this certificate in the exercise of its authority over the licensing of operators.

(Date) | (Facility) | (Signature of applicant)

SIGN YOUR NAME IN INK AS IT APPEARS ON YOUR APPLICATION FOR OPERATOR'S OR SENIOR OPERATOR'S LICENSE

MEDICAL EXAMINATION

39. **Doctor:** It is **essential** that **each of the items on this page be completed. Sign** the Certificate and **mail to** the Director, Division of Reactor Licensing, United States Nuclear Regulatory Commission, Washington, D.C. 20555 **for reactor** operator license applicants or to the Director, Division of Materials and Fuel Cycle Facility Licensing, United States Nuclear Regulatory Commission, Washington, D.C. 20555 for applicants for **non-reactor** licenses.

40. Physician's Summary and elaboration of the medical history on front of report. Use additional sheet if more space is needed.

Physical Examination. Give details of abnormal findings under item 20 below.

1. Date of examination		2. Height	3. Weight
4. Blood pressure		5. Pulse	
6. Distant visual acuity uncorrected	right	left	method used
*7. Distant visual acuity corrected	right	left	(data required only if corrective lenses are normally worn)
8. Near visual acuity uncorrected	right	left	method used
*9. Near visual acuity corrected	right	left	(data required only if corrective lenses are normally worn)
10. Color vision			method used
11. Gross visual fields			
12. Hearing	right	left	method used
13. Eyes, general		14. Pupils	
15. Ophthalmoscopic			
16. Ears, general		17. Drums	
18. Heart		19. Vascular system	

20. Details and evaluation of any item 1 through 19, above, reported abnormal and summary evaluation of over-all condition.

21. Did the foregoing examination reveal any mental or physical disability which might cause impaired judgment or motor coordination? ☐ YES ☐ NO

I understand that any of the information in this examination may be used by the Nuclear Regulartory Commission in the exercise of its authority over the licensing of operators.

(Date)

(Signature of examining physician)

DOCTOR: IT IS REQUIRED THAT EVERY ITEM ON THIS PAGE BE COMPLETED EXCEPT THOSE MARKED WITH * WHEN NOT APPLICABLE.

Typed or printed name of examining physician

Address

State in which licensed

CERTIFICATE OF MEDICAL EXAMINATION

Full and accurate completion of NRC Form-396 is essential to the expeditious processing of applications. Incomplete or unclear medical reports will cause delay and requests for further information. When significant abnormalities exist, the detailed reporting of manifestations, diagnoses, and prognoses with supporting data may expedite the processing of applications and eliminate the need for additional correspondence. Special attention is directed to the following items in the Physical Examination section of the report form:

A. Methods, where requested, must be indicated.

B. Visual acuity (Items 6-9) should be reported in a conventional manner. Near-point and far-point designations are not a satisfactory substitute for reports of ACUITY.

C. If corrective lenses are NOT worn, there is no need to complete lines 7 or 9. If corrective lenses are worn, all four items -- 6, 7, 8 and 9 -- must be completed.

D. Visual fields (Item 11) must be reported in degrees or as "full," "complete," or "normal" (or equivalent).

E. If hearing is not normal (Item 12), audiometric data must be reported. (Refer to Item H below.)

F. An indication of "not applicable," "no significant information," or other similar notation should be entered for items where such a notation is appropriate, instead of merely failing to make an entry for such items.

G. Completion of all items, including Item 21, is required, except as indicated in C. above.

H. If color vision (Item 10) or hearing (Item 12) is not normal, results of a practical test should be submitted indicating the degree to which the lack of color vision or diminished hearing interferes with the individual's ability to perform licensed duties.

APPENDIX E

Typical Sample Questions
Operator and Senior Operator Examinations

This appendix contains lists of sample questions typical of those appearing in operator and senior operator written examinations. The questions are grouped within the categories described in this guide, and effort has been made to indicate which questions apply solely to specific reactor types, such as power, test or research reactors.

It should be noted that some appropriate operator or senior operator questions are not listed herein in either the operator or senior operator categories because operator or senior operator duties would not necessitate knowledge of the subject being investigated by the question. For example, at some power reactors, the management has determined that calculations involving determination of control rod worths are sufficiently complex to necessitate that only the plant reactor engineer be responsible for performing the calculations, although the senior operator on shift directs the activities of the licensed operators.

Conversely, some typical senior operator questions given may be investigating areas in which operators at some facilities need a higher level of knowledge because of the nature of their assigned duties; thus, questions of this type might be included in an operator written examination at such a facility. As an example of this aspect, at a low power research reactor, calculations involving control of coolant chemistry are quite simple and straightforward, and operators can and do make such calculations when Management so directs.

Reference will be made to certain types of reactors throughout this appendix. For the purpose of this guide, some definitions of these types are in order for consistency and clarification, and are as follows:

1. POWER REACTORS are those utilized to produce steam primarily in order to drive turbine generators for the production of electricity or providing a means of propulsion. Reactor powers are generally in excess of 100 MW(t).

2. TEST REACTORS are those utilized for materials testing and isotope production; reactor powers are generally in excess of 5 MW(t).

3. RESEARCH AND TRAINING REACTORS -- self-explanatory; reactor powers --large, generally 100 KW(t) - 5 MW(t); small, generally <100 KW(t).

4. CRITICAL FACILITIES are those reactors utilized for critical experiments and training; reactor powers -- generally $\leq$ KW(t).

I. OPERATOR EXAMINATION

A. PRINCIPLES OF REACTOR OPERATION

In the list that follows, the asterisked questions are types that would usually only appear in examinations given at facilities with reactors operating at power levels greater than 100 KW. Also, questions such as 7 and 8 would only be appropriate for water reactors and reactors with cores containing fertile material, respectively.

1. Briefly explain the following:

   a. Reactor period

   b. Reflector

   c. Reactivity

   d. Criticality

   e. Neutron flux

   f. Nuclear cross-section

   g. Prompt critical

   h. Subcritical multiplication

   *i. Doppler coefficient

2. a. What is the significance of delayed neutrons in reactor control?

   b. What is the origin of these neutrons?

3. a. What is the function of the neutron source installed in the reactor?

   *b. What nuclear characteristics of the reactor core will enable the operator to start up the reactor without this source once considerable high power operation has been logged?

4. *a. What is meant by "xenon poisoning?"

*b. In an operating condition with equilibrium xenon in the reactor, there are certain factors which collectively create this "equilibrium" condition. What are they and how does the magnitude of each change if the reactor power level is quickly lowered and leveled off?

5. As operator, you are bringing the reactor to power on a stable period of 30 seconds. At a certain time the nuclear instrumentation indicates a level of 10 watts. What level will be indicated at the following times:

   a. two minutes?

   b. three minutes?

6. What nuclear phenomenon contributes the largest amount of heat generation:

   a. During the fission process?

   *b. Two hours after the reactor has been shutdown from full power?

7. One frequently hears about "nitrogen-16" associated with water-cooled reactors. What is this, and of what significance is it to the operators?

*8. What effects are primarily responsible for the ability of the reactor core to breed?

B. FEATURES OF FACILITY DESIGN

All of the questions in this list are typical of those that are, in general, appropriate for most reactors that employ a liquid moderator or coolant to surround the core; however, the following comments are appropriate:

1. Questions 3, 6 and 8 would only apply at a power reactor.

2. Schematic detail in Questions 4 and 5 will depend upon reactor size and purpose.

In addition, Question 4 would not be appropriate for a reactor core that utilized only convection cooling and Question 5(b) would not apply to an open pool or tank type reactor facility.

1. Sketch a plan view of the reactor core showing the location of fuel elements, control rods, reflector, irradiation facilities, and neutron detectors.

2. Sketch a fuel element, including the most important dimensions.

3. Describe the flow cycle within the reactor vessel from entry of feedwater until steam is discharged to the turbine.

4. Draw a simplified schematic diagram of the primary coolant system, showing all major components and vital instrumentation. Include flow directions.

5. Sketch flow diagrams for:

   a. The purification system.

   b. The pressure control and relief system.

   Label the major components (pumps, valves, tanks, etc.), show flow directions, and locate all temperature, pressure, flow and level detectors.

6. List the components that are cooled by the intermediate cooling system. Why is such a "buffer" system used instead of river water?

7. Briefly describe the shielding provided at the facility, starting with the reactor core and moving radially outward.

8. a. Describe the reactor coolant pump seal arrangement. Include flow direction, flow rates and where any leak-off goes.

   b. Describe how a seal failure would be detected.

C. GENERAL OPERATING CHARACTERISTICS

This list contains types of questions which pertain to the operating characteristics of specific reactors and auxiliary systems, but are not generally applicable as are those questions listed in Category A. Specifically, Questions 1, 2 and 7 would be most appropriate for power reactor examinations, Questions 4 and 8 primarily appropriate for large research and test reactors, and Question 6 generally would apply only at small research reactors or critical facilities.

Question 3 would only apply at facilities where the reactor moderator or coolant would undergo a considerable temperature swing during operation.

Question 5 is a general type, with only the power level specified in 5(b) being a variable, depending upon the reactor's power range.

1. Describe how the pressurizer functions to limit pressure variations during step load increases and decreases.

2. State whether each of the following occurrences tend to increase or decrease reactivity, assuming that no scrams occur:

a. Feedwater temperature decrease during normal full power operation

b. Turbine trip at full power

c. Loss of power to one recirculation pump -- reactor at full power

3. Does the worth of a control rod change from hot operating conditions to cold operating conditions? Explain your answer.

4. The reactor is started and placed on automatic operation after reaching one megawatt with the regulating rod halfway withdrawn. With no other changes to the reactor or the position of the safety rods, describe the position change of the regulating rod as a function of time.

5. For the operation specified, sketch the traces you would expect to see on a recorder connected to the following nuclear instrumentation channels:

   a. Startup Channel during rod withdrawal of one increment followed by a one minute pause; K_{eff} after withdrawal = 0.996.

   b. Log N Channel during scram from 20 MW(t). Show trace to 10 minutes following the scram.

6. Assume the reactor is critical at a level of 1 watt. What effect, if any, will be observed if the source is removed at this point?

7. What methods are used to control oxygen concentration in the primary and secondary system and what concentrations are the upper limits during normal reactor operation.

8. What are the fuel cycle reactivity requirements for the following:

   a. Cold critical to hot critical

   b. Equilibrium xenon

   c. Fuel depletion

   d. Allowance for xenon override.

D. INSTRUMENTS AND CONTROLS

Many of the questions given for this category; i.e., 1, 2, 5, 6 and 7, may be appropriate for all types of reactors; however, questions 4 and 8 apply only to power reactors, while question 3 again would be a type normally applicable for reactors with power levels greater than 100 KW.

1. Sketch block diagrams of the following nuclear instrumentation channels.

   a. Startup Channel

   b. Log N Channel

   c. Power Range Channel

2. For each nuclear detector in the nuclear instrumentation channels, discuss the following:

   a. Operating principle of the detector

   b. Method used to distinguish neutrons from other radiation.

3. Sketch a compensated ion chamber recorder trace following a scram for the following situations. Start all curves at 100% on the Log N recorder. Indicate on the traces the reasons for all significant slope changes.

   a. CIC Perfect compensation, no power history

   b. CIC Perfect compensation, many MW days operation

   c. CIC Undercompensated, many MW days operation

   d. CIC Grossly overcompensated, no power history.

4. Describe the sequential operation of the automatic plant control system for a load increase demand, starting with turbine throttle valve opening.

5. Briefly describe the operation of the automatic regulating rod control system. A block diagram may be used for clarity.

6. What interlocks must be satisfied before the control rods can be withdrawn for a reactor startup?

7. How is the temperature of the purification water controlled?

8. Discuss in general terms the in-core flux monitoring system, telling why it is needed, what information it gives you, and what protection it affords.

E. SAFETY AND EMERGENCY SYSTEMS

This listing includes eight questions which may appropriately appear in an examination at any type of reactor facility, small or large. Only the nomenclature would differ for given facilities; e.g., the safety amplifier in Question 2 might be another type of electronic tripping device. In addition, the engineered safeguards system referred to in Question 7 and the containment of Question 3 would often not exist at a critical facility or small research reactor.

1. List the parameters, systems or conditions which will produce automatic reactor shutdown and give the set points at which scrams will occur.

2. Explain briefly the operation of the magnet safety amplifier during a scram condition.

3. What signals isolate the containment building?

4. Since the protection system provides scram action, why does it also have a setback action?

5. a. Sketch a simplified schematic diagram and briefly describe the operation of the emergency "poison" system.

   b. What does the "poison" consist of and how great a reactivity increase will it compensate for?

6. List the equipment that is on the circuit supplied by the emergency diesel generator.

7. a. Sketch the Low Pressure Safety Injection system and show major components and valves.

   b. Describe the changes that take place throughout the engineered safeguards systems upon receipt of a Recirculation Actuation Signal.

8. Explain how the possibility of scram system failure is reduced by design.

F. STANDARD AND EMERGENCY OPERATING PROCEDURES

In this listing, only Questions 1.a, 1.b, 4, 6 and 7 are types generally applicable to all classes of reactor facilities. Questions 2, 5 and 8 would be most appropriate for a test reactor or power reactor, while Question 3 is most applicable to a research or test reactor only.

1. List the following provisions of your operating procedures:

   a. Minimum reactor period while increasing power.

   b. Maximum activity of the reactor coolant.

   c. Maximum heating rate for reactor vessel.

2. Briefly outline the MAJOR steps that will be followed in bringing the reactor from full power to cold shutdown.

3. Describe the procedural requirements which must be met before you, as an operator, would insert a sample into or near the reactor for irradiation.

4. Assume that, during a startup, the reactor becomes critical at a significantly lower control rod position than expected. What action should be taken by the operator?

5. While operating at 60 MW(t), reactor power drops suddenly (indicated by ALL power monitors) to 30 MW(t) and begins to level off. What should be your actions as reactor operator on duty? What might have caused this?

6. Assume the reactor is not being operated, (i.e., no control rods withdrawn) but that the control power key is in the switch. What control room personnel requirements must be met?

7. After an extended shutdown, the reactor has just returned to power when an electrical outage occurs. Must the nuclear instrumentation be recalibrated before starting up the reactor? Explain.

8. What immediate action would you take if you received word of a 3 GPM leak in the primary cooling system while you were operating the reactor?

G. RADIATION CONTROL AND SAFETY

All questions listed in this category are general types that are usually appropriate at all types of reactor facilities.

1. Provide values for the following:

   a. The maximum permissible quarterly radiation dose to the whole body.

   b. The maximum quarterly radiation dose allowed for hands.

   c. The maximum quarterly radiation dose allowed for the skin.

   d. The maximum and minimum radiation levels expected in an area marked "Caution, Radiation Area."

   e. The maximum and minimum radiation levels expected in an area marked "Caution, High Radiation Area."

2. Name four types of radiation that could be present in reactor buildings, and list them in decreasing order of biological damage, assuming 1 RAD of each type of radiation.

3. Is there any difference between radiation and contamination? Explain your answer.

4. What is a "Radiation Work Permit?" What items must be listed on such a permit? What conditions terminate an RWP?

5. Tell the following about the Fixed Radiation Monitoring System:

   a. Type of detectors employed.

   b. Types of radiation measured by each of these detectors.

   c. Why is neutron monitoring considered necessary in the spent fuel storage areas?

6. List the portable radiation monitoring equipment available. Indicate the type or types of radiation detected by each instrument.

7. If the radiation intensity is 3.2 Rem/hr at two feet from a small source, what will the intensity be 8 feet from the source?

8. What action would you take after each of the following events occur?

   a. While working with contaminated material, a sharp edge cuts your hand.

   b. While drawing a sample from the purification system, a line parts and you are sprayed with water.

   c. While walking by the Hand and Foot monitor, you notice a high reading on the meter.

II. SENIOR OPERATOR EXAMINATION

H. REACTOR THEORY

In this grouping of typical questions, Nos. 1 thru 4, 7a and 7c would generally be appropriate for all types of reactor facilities, although the answer to Question 3 should contain more explanation for power and test reactors; i.e., to consider the effects of burnup, temperature swing, etc.

Questions 5, 6, 7b, 7d and 8 are usually only appropriate for power or test reactors with Question 5 specifically applicable to a boiling water reactor; however, this type of question could be made appropriate for any power reactor, if slight modifications are made in the terminology used.

1. Assume 1000 neutrons are born from thermal fission. List, in sequence, what can happen to these neutrons. Approximately how many would have to cause fission to sustain a chain reaction?

2. A reactor is operating at a just critical level of 100 watts.

   a. If a 0.002 ΔK step is added, approximately what will the resulting stable period be?

   b. Neglecting the effects of delayed neutrons, approximately what period would be obtained for the step increase in (a)?

3. Discuss how and why the worth of an individual control rod can be variable throughout reactor operation. Include both axial and radial effects in your discussion.

4. A shutdown reactor has a source range reading of 270 CPM. The operator begins to pull one of three supposedly equal worth rods. The count rate rises to 450 CPM when the rod is completely withdrawn. The operator now pulls the second of the three rods and the count rate rises to 1350 CPM when the rod is completely withdrawn. Does the much greater increase in count rate show that the rods are not really equal or is this to be expected? Explain and demonstrate your answer.

5. Discuss how the following factors change, if any, in a boiling water reactor during an increase in power:

   a. Resonance absorption.

   b. Neutron leakage.

   c. Thermal utilization.

6. Compare the reactivity effects of xenon poisoning to that of samarium. Include transient and long-term steady-state effects in your discussion.

7. Define: a. Isotope

   b. Burnout heat flux

   c. Neutron lifetime

   d. Core conversion ratio

8. a. Define the term "nucleate boiling."

   b. When this type of boiling occurs, how and why is the heat transfer coefficient affected?

I. RADIOACTIVE MATERIALS HANDLING, DISPOSAL & HAZARDS

In this section, Questions 1, 2, 5, 6, 7 and 8 are generally applicable to all facilities, with the depth of detail for the answers to 1 and 2 depending upon reactor size and purpose.

Question 3 would be primarily appropriate for power or test reactors, while Question 4 illustrates how the same general type of question is appropriate at large research facilities.

1. Sketch a block diagram of the liquid waste disposal system. List the plant drains which go directly to the tanks, and indicate direction of flow and capacity of the tanks. Omit lines below 1 inch diameter.

2. Discuss how gaseous waste is monitored, collected and disposed of at your facility, and list the normal sources of such gases.

3. As senior operator on shift, you are notified that a primary coolant sample has yielded a radioactivity level of 10 times higher than the previously taken sample normal for 20 MW(t) operation:

   a. What is the activity level of the high sample?

   b. What corrective action, if any, should be taken?

   c. How could it be determined whether the sudden increase in activity was due to a fuel element rupture or a "crud burst?"

4. With the reactor operating at 1 MW(t), one of the fuel elements ruptures.

   a. How would the oeprator first receive an indication of the failure?

   b. As senior operator on duty, what should be your immediate actions upon suspecting the element has failed?

   c. Outline the methods that should be used to remove the element and clean up the pool.

5. What is the approximate attenuation to a 1 Mev gamma rays afforded by:

   a. One inch of lead shielding?

   b. One foot of water shielding?

   μ of lead = 2.3/inch

   μ of water = 0.19/inch.

6. An operator in his haste to run from a suddently exposed gamma source, trips and strikes his head. Assuming that he lay unconscious three feet from the source, calculate the following:

   a. Average whole body dose received in one hour.

   b. If the man were not removed for eight hours, discuss the probable state of his health after the exposure.

   Note: Source strength is 20 curies, emitting gamma energies of 1.7 and 1.4 MEV.

7. What factors should be considered by the shift supervisor when he assigns jobs that involve radiation exposure?

8. Assume there is a release of radioactive material within the reactor building and it is necessary that your operators enter the building area without waiting for a health physicist to arrive.

   a. What kind of monitoring equipment would you require?

   b. What precautions would you take prior to entry?

   c. Explain in detail how you would determine how long the operators could stay in the vessel.

J. SPECIFIC OPERATING CHARACTERISTICS

Typical questions similar to Nos. 4, 5, and 7 would be appropriate for all reactor facilities.

Questions 2 and 8 generally are only appropriate for power reactors, although Question 8 could be modified for use on a large research or test reactor examination.

Question 3 is specifically applicable to a pressurized water reactor.

Questions 6 and 7(b) are appropriate for either power, test, or large research reactors.

As stated in Part V of the guide, for typical questions involving interpretation and use of graphs or curves, the curves would be distributed along with the examination. Thus, the appropriate curves would accompany Question 8.

1. Sketch graphs that show the response with time of the following variables following a turbine trip. Assume the plant was at 100% power on fully automatic control and that no operator action is taken following the trip:

   a. Reactor power.

   b. Reactor outlet temperature.

   c. Reactor inlet temperature.

   d. Steam outlet temperature.

   e. Steam flow (per loop).

2. On your answer sheet, list the following ranges of values for normal power operation, including units, in tabular form.

	Primary Coolant	Intermediate Coolant
Max. Total solids	_____	_____
Max. dissolved solids	_____	_____
Max. chloride	_____	_____
Dissolved hydrogen	_____	_____
ph	_____	_____

3. The reactor power is 70%; Group D control rods are at 210 steps and in manual; the boron concentration is 510 ppm. This power level has been maintained for two days.

   a. Calculate the quantity (in gallons) of water and/or boric acid necessary to increase the power level to 100%. Show all work and mark all curves to indicate where you got your information (See Figures J1 - J9).

   b. The power is to remain at 100% for some time. Calculate the ppm change in boron necessary to maintain proper operation over the next 2 days.

   c. Describe how the operator will know when and how fast to add water and/or boric acid to maintain proper operation.

   d. How and why will the axial flux imbalance change when going from 70% to 100% power?

4. The reactor is critical at 50 watts of power and increasing on a stable reactor period. The startup channels show a 100 second period and the intermediate channels show a 65 second period. Which period do you consider is the more accurate? Explain your answer.

5. Outline a method that may be used to determine the cold minimum core shutdown margin during low reactor power operation.

6. When the axial flux shape is plotted after one day's operation at full power and again after 21 days full power operation, the maximum flux level and location is different for the two cases.

   a. Will the maximum flux be located higher or lower in the core at the later date?

   b. Explain your answer.

7. a. How are the power range nuclear detectors calibrated?

   b. Using typical full power values for the parameters needed, calculate the reactor thermal power.

8. Assume the reactor has been at full power, 600 MW(t), for two months. Controlling Rod Group "A" is banked at 27". A full scram occurs, the cause of the scram has been determined, the situation remedied and the control rods are to be withdrawn. Calculate the just critical control rod bank position necessary to reach a power level adequate to commence nuclear heating.

K. FUEL HANDLING & CORE PARAMETERS

In this grouping, Questions 2, 4(a), 5, 6, and 7 are appropriate for all reactor facilities, except that the size and amount of fuel handling equipment referred to in Question 6 will be quite variable, e.g., at a small research reactor, the fuel handling apparatus may consist of a hook tool and a rope snare.

Questions 1, 3, 4(b) and 8 apply primarily to power reactors, with possible application of 4(b) to large research and test reactors.

1. During fuel loading the records become confused so that the enrichment of a new fuel element is not known for sure. Where and what type of identification is on the fuel element to indicate the enrichment of the element?

2. As the water to metal ratio of a reactor core is increased over that which is optimum, reactivity decreases due to an increasing "overmoderation" condition. Explain WHY this phenomenon causes a reactivity decrease, past the optimum point.

3. During a normal plant heatup from 250°F, the reactor is maintained at 1×10^{-7} amps on channel 5 to produce a heatup rate of 35°F/hr. Will the same indicated power produce the same heatup rate as the plant approaches operating condition? Explain your answer.

4. a. List and give the half-lives of installed sources in the reactor core, and explain how each source produces neutrons.

   b. Explain how the nuclear characteristics of the Antimony-Beryllium source enable it to be termed "regenerative".

5. A startup detector is placed near a reactor core to monitor a new core loading. As the loading proceeds, the following data is obtained.

Step	No. of Elements	CPS **
**	0	10
1	4	12
2	8	15
3	10	20
4	12	30
5	13	60
6	14	150

a. Estimate the number of fuel elements that will make the reactor critical and show how you arrived at this figure.

b. Do you think the detector is optimumly placed, too close or too far away from the source? Explain.

6. List the major equipment which comprises the fuel handling system, and briefly describe the purpose of each.

7. What procedural requirements must be met before control rod or fuel followers may be transferred to or from the core?

8. a. What is the purpose of control rod programming?

b. Explain why the fuel enrichments in certain sections of the core differ from each other.

L. ADMINISTRATIVE PROCEDURES, CONDITIONS & LIMITATIONS

In this section, Questions 4, 5, 6 and 8 are usually appropriate for all reactor facilities.

Questions 1, 2 and 3 are generally applicable for power reactors, while Question 7 would be most appropriate for research and test reactors.

1. There are 3 types of tags used at your facility: Danger, Caution and Hold For Inspection.

a. Indicate when each type of tag would be used.

b. What is the procedure for "tagging clearance"?

2. List the following administrative limitations:

a. Maximum rate of generator load increase or decrease.

b. Containment building maximum internal pressure.

c. Maximum reactor power with vessel head removed.

3. Assume that you are shift supervisor and no higher supervision is at the plant. The reactor is operating at full power and a scram of undetermined origin occurs. The system dispatcher informs you that the power is badly needed on the line. How long would you wait before ordering the reactor to be started up?

4. Which individuals, by position title, in the plant organization may authorize the following operations?

a. Adjusting the discrimination in the startup channel.

b. Disposing of liquid waste from the site.

c. Scramming the reactor.

5. Assume that you are the senior operator on duty and are supervising a core loading change. An operator is lowering a fuel element into the core (it is halfway inserted) when the reactor top fixed radiation monitor alarms. What should be your actions?

6. Under what conditions must a proposed change to a reactor operating procedure be referred to the Safeguards Committee for review?

7. List the types of experiments which will NOT be accepted for insertion in the core region, according to the technical specifications.

8. On your answer sheet, complete the following table by listing the personnel that MUST be in each status for the event given.

Event	In Control Room	At Facility	On Call
a. Replacement of fuel in core.----------	_____	_____	____
b. Reactor re-start after scram from high flux.--	_____	_____	____
c. Reactor power escalation after a 10% power drop, cause unknown. Conditions appear normal.----------	_____	_____	____
d. Normal reactor shutdown.----------	_____	_____	____

APPENDIX F

Eligibility for Examination With No Reactor Startup Demonstration

Normally, a reactor startup is required as part of the operating test administered at the facility. However, an alternative program, described below, may be completed for power reactor applicants in which case no actual reactor startup needs to be performed during the license examination.

REQUIREMENTS TO BECOME ELIGIBLE FOR EXAMINATION AT AN OPERATING POWER REACTOR WITH NO REACTOR STARTUP DEMONSTRATION.

A. Educational Requirements

As specified in ANSI N18.1-1971.

B. Lectures

A total of 500 hours on subjects listed in ANSI N18.1-1971, Section 5.2.1, related subjects and prerequisite courses. These lectures are in addition to the lectures in the one week training program in Section D below.

C. Power Plant Experience

Applicants shall meet the following requirements:

1. ANSI N18.1-1971, Section 4.5.1 or 4.3.1 as applicable.

2. A minimum time of six months at the site. At least three months of this time shall include participation in on-the-job training which involves manipulation of the nuclear power plant controls during day to day operation. This phase of the training will be programmed and supervised. A record of training accomplishment should be maintained and signed off by the training coordinator or other responsible individual.

3. Applicants must participate in reactor and plant operation at facility power levels up to at least 20% power prior to the examination.

4. The applicant has manipulated the controls of the reactor facility during five significant reactivity changes as described in the operator requalification program for the facility. Every effort should be made to have a diversification of reactivity changes for each applicant.

D. Simulator Training

The applicant has satisfactorily completed a NRC approved training program of at least one week duration at a nuclear power plant simulator. The application shall contain a certification from the simulator training center attesting to the applicant's:

1. ability to manipulate the controls and keep the reactor under control during a reactor startup,

2. ability to predict instrument response and use the instrumentation during a reactor startup,

3. ability to follow the facility startup procedures, and

4. ability to explain alarms and annunciators that may occur during this operation.

E. Review and Evaluation

A minimum of 40 hours for review, audit examinations and evaluation of the applicant should conclude the training program.

F. Records

Documentation shall be maintained as specified in ANSI N18.1-1971, Section 5.6.

APPENDIX G

NRC FORM 157
(10-75)

U.S. NUCLEAR REGULATORY COMMISSION

EXAMINATION REPORT

LICENSE APPLIED FOR: RO ☐ SRO ☐

CURRENTLY A LICENSED OPERATOR HERE? YES ☐ NO ☐

APPLICANT'S NAME	REACTOR	LOCATION

WRITTEN EXAMINATION

OPERATOR (SECTION 55.21)	ADMINISTERED BY:	DATE	WAIVED ☐
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES A____% B____% C____% D____% E____% F____% G____%		FAILED ☐
SENIOR OPERATOR (SECTION 55.22)	ADMINISTERED BY:	DATE	WAIVED ☐
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES H____% I____% J____% K____% L____%		FAILED ☐

OPERATING TEST

(SECTION 55.23)	ADMINISTERED BY:	DATE	WAIVED ☐ PASSED ☐ FAILED ☐
			HOT ☐ COLD ☐

COMMENTS

RECOMMENDATION	APPROVE FOR SENIOR LICENSE ☐ OPERATOR LICENSE ☐	SIGNATURE
	DO NOT APPROVE FOR SENIOR LICENSE ☐ OPERATOR LICENSE ☐	EXAMINER DATE
ISSUE LICENSE	SENIOR LICENSE ☐ OPERATOR LICENSE ☐	SIGNATURE
		Chief, Operator Licensing Branch

NOTE: The operator and oral examination check sheet must be completed for each applicant. If applicant has failed the operating and oral test at the Senior Level, the check sheet should be completed to indicate applicant's evaluation at the operator level.

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(10-75)

OPERATING AND ORAL EXAMINATION SUMMARY REPORT

If an SRO applicant is unsatisfactory at the SRO Level, complete the RO column to evaluate applicant's adequacy at the RO Level. **For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.*	EVALUATION		
	SRO	RO	*PAGE NUMBER FOR "U"
1. OPERATING DEMONSTRATION			
1.1 Pre-startup and Instrument Checks			
1.2 Console Operation			
a. Manipulations			
b. Understanding			
2. FACILITY EQUIPMENT			
a. Major			
b. Auxiliary			
c. Engineered Safeguards Systems			
3. INSTRUMENTATION			
a. Nuclear			
b. Process			
4. REACTOR PROTECTION			
5. PROCEDURES			
a. Normal			
b. Abnormal			
c. Emergency			
6. REACTIVITY EFFECTS (Except Console Operation)			
7. ADMINISTRATIVE REQUIREMENTS			
8. EMERGENCY PLAN			
9. RADIATION PROTECTIO			
10. RESPONSIBILITIES			

COMMENTS

NRC FORM 157
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OPERATING AND ORAL EXAMINATION NOTES

A. OPERATING DEMONSTRATION	EVALUATION
1.1 Pre-startup and Instrument Checks	
Type of checkout: *(specify)*_____	
1.1.1 Familiarity with check sheet	
1.1.2 Accuracy when reading instruments	
1.1.3 Understanding of what is being checked	
1.1.4 Understanding of reasons for checkout	
1.1.5 Effects of malfunctions	
1.1.6 Knowledge of Control Room reference data	
1.1.7 Method of ECP determination	
1.2 Console Operation	
a. Initial conditions:	
b. Program:	
1.2.1 Follows procedures	
1.2.2 Ability to predict response for specified program	
1.2.3 Observes and checks instrumentation	
1.2.4 Understanding of instrument response	
1.2.5 Ability to follow specified program accurately	
1.2.6 Dexterity and "feel" for console controls	
1.2.7 Knowledge of reactivity effects	

COMMENTS: *(Required for "U")*

NRC FORM 157
(10-75)

B. CONTROL ROOM
(Major, Auxiliary and Engineered Safeguards Systems)

	SYSTEMS							
	A	B	C	D	E	F	G	H
2.0 EQUIPMENT								
2.1 Purpose								
2.2 Flow Path								
2.3 Normal Parameters								
2.4 Components								
2.5 System Behavior and Response								
3.0 INSTRUMENTATION								
3.1 Detector								
3.2 Malfunction								
3.3 Control Room Indication								
4.0 REACTOR PROTECTION								
4.1 Alarms/Setpoints								
4.2 Safety System Input								
4.3 Interlocks								
5.0 PROCEDURES								
5.1 Normal Procedures								
5.2 Abnormal Procedures								
5.3 Emergency Procedures								
6.0 REACTIVITY EFFECTS								
7.0 ADMINISTRATIVE REQUIREMENTS								
7.1 Technical Specif ıtions								
7.2 Facility Requirements								

COMMENTS: *(Required for "U")*

☐ CONTINUED ON REVERSE

NRC FORM 157
(10-75)

B. CONTROL ROOM
(Nuclear and Radiation Instruments)

	SYSTEMS				
	A	B	C	D	E
3.0 INSTRUMENTS					
3.1 Detectors					
3.2 Malfunctions					
3.3 Control Room Indications					
3.4 Channel Components					
3.5 Compensation/Discriminator					
3.6 Input to Control System					
4.0 REACTOR PROTECTION					
4.1 Alarms/Setpoints					
4.2 Safety System Input					
4.3 Interlocks					
5.0 PROCEDURES					
5.1 Normal Procedures					
5.2 Abnormal Procedures					
5.3 Emergency Procedure					
7.0 ADMINISTRATIVE REQUIREMENTS					
7.1 Technical Specifications					
7.2 Facility Requirements					

COMMENTS: *(Required for "U")*

NRC FORM 157
(10-75)

B. CONTROL ROOM
(Electrical)

	SYSTEMS			
	A	B	C	D
2.0 EQUIPMENT				
2.1 Purpose				
2.2 Flow Path				
2.3 Normal Parameters				
2.4 Components				
2.5 System Behavior or Response				
3.0 INSTRUMENTS				
3.2 Interlocks				
3.4 Control Room Indication				
5.0 PROCEDURES				
5.1 Normal Procedures				
5.2 Abnormal Procedures				
5.3 Emergency Procedures				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS: *(Required for "U")*

C. REACTOR AND AUXILIARY BUILDINGS

(Major, Auxiliary, Electrical Safeguards, Fuel Handling, Rad Waste)

	SYSTEMS					
	A	B	C	D	E	F
2.0 EQUIPMENT						
2.2 Flow Paths						
2.3 Normal Parameters						
2.4 Equipment Location						
2.5 System Behavior and Response						
3.0 INSTRUMENTS						
3.8 Local Instrumentation						
5.0 PROCEDURES						
5.1 Normal Procedures (Local)						
5.2 Abnormal Procedures (Local)						
5.3 Emergency Procedures (Local)						
6.0 REACTIVITY EFFECTS						
7.0 ADMINISTRATIVE REQUIREMENTS						
7.1 Technical Specifications						
7.2 Facility Requirements						
9.0 RADIATION PROTECTION						
9.1 Radiation Control Procedures						
9.2 Portable Instruments – Location						
9.3 Portable Instruments – Use						
9.4 Portable Instruments – Characteristics						
9.5 Personal Practices						
8.0 EMERGENCY PLAN						
8.1 Equipment Locations						
8.2 Equipment Use						
8.3 Duties						

COMMENTS *(Required for "U")*

NRC FORM 157
(10-75)

D. DISCUSSIONS *(Integrated Plant Response)*

	SYSTEMS			
	A	B	C	D
2.0 EQUIPMENT				
2.6 Components Response				
3.0 INSTRUMENTS				
3.4 Control Room Indications				
3.8 Automatic Control				
3.9 Ability to Manipulate Manual Control				
4.0 REACTOR PROTECTION				
4.1 Automatic Actions				
4.2 Alarm/Setpoints				
5.0 PROCEDURES				
5.1 Normal Procedures				
5.2 Abnormal Procedures				
5.3 Emergency Procedures				
6.0 REACTIVITY EFFECTS				
6.3 Coefficient Effects				
6.6 Transient Analysis				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS *(Required for "U")*

D. DISCUSSION	EVAL-UATION
6.0 REACTIVITY EFFECTS *(Nuclear Theory)*	
6.1 Subcritical Multiplication	
6.2 Delayed Neutrons Effect	
6.3 Coefficients	
6.4 Poison Effects	
6.5 Long Term Exposure Effects	
9.0 RADIATION PROTECTION	
9.1 Radiation Control Procedures	
9.4 Source and Types of Radiation	
9.6 Contamination	
9.7 Exposure Limits *(10 CFR 20) (Facility)*	
10.0 RESPONSIBILITY (QA, NON-RAD SAFETY, AP's, SECURITY)	

COMMENTS *(Required for "U")*