# Proposed Rules

**Federal Register**

Vol. 64, No. 47

Thursday, March 11, 1999

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

## NUCLEAR REGULATORY COMMISSION

### 10 CFR Parts 21, 50, and 54

**RIN 3150–AG12**

### Use of Alternative Source Terms at Operating Reactors

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Proposed rule.

**SUMMARY:** The Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to allow holders of operating licenses for nuclear power plants to voluntarily replace the traditional source term used in design basis accident analyses with alternative source terms. This action would allow interested licensees to pursue cost beneficial licensing actions to reduce unnecessary regulatory burden without compromising the margin of safety of the facility. The NRC is also proposing to amend its regulations to revise certain sections to conform with the final rule published on December 11, 1996, concerning reactor site criteria.

**DATES:** The comment period expires on May 25, 1999. Comments received after this date will be considered, if it is practical to do so, but the NRC is able to assure consideration only for comments received on or before this date.

**ADDRESSES:** Mail written comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, Mail Stop O16C1.

Deliver comments to: One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

You may also submit comments via the NRC's interactive rulemaking web site, ''Rulemaking Forum,'' through the NRC home page (http://www.nrc.gov). This site enables people to transmit comments as files (in any format, but WordPerfect version 6.1 is preferred), if your web browser supports that function. Information on the use of the Rulemaking Forum is available on the website. For additional assistance on the use of the interactive rulemaking site, contact Ms. Carol Gallagher, telephone: 301–415–5905; or by Internet electronic mail to cag@nrc.gov.

Certain documents related to this rulemaking, including comments received and the environmental assessment and finding of no significant impact may be examined at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. These same documents also may be viewed and downloaded electronically via the interactive rulemaking website established by NRC for this rulemaking.

**FOR FURTHER INFORMATION CONTACT:** Mr. Stephen F. LaVie, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone: (301) 415–1081; or by Internet electronic mail to sfl@nrc.gov.

**SUPPLEMENTARY INFORMATION:**



## I. Background

A holder of an operating license (i.e., the licensee) for a light-water power reactor is required by regulations issued by the NRC (or its predecessor, the U.S. Atomic Energy Commission, (AEC)) to submit a safety analysis report that contains assessments of the radiological consequences of potential accidents and an evaluation of the proposed facility site. The NRC uses this information in its evaluation of the suitability of the reactor design and the proposed site as required by its regulations contained in 10 CFR Parts 50 and 100. Section 100.11, which was adopted by the AEC in 1962 (27 FR 3509; April 12, 1962), requires an applicant to assume (1) a fission product release from the reactor core, (2) the expected containment leak rate, and (3) the site meteorological conditions to establish an exclusion area and a low population zone. This fission product release is based on a major accident that would result in substantial release of appreciable quantities of fission products from the core to the containment atmosphere. A note to § 100.11 states that Technical Information Document (TID) 14844, ''Calculation of Distance Factors for Power and Test Reactors,'' may be used as a source of guidance in developing the exclusion area, the low population zone, and the population center distance.

The fission product release from the reactor core into containment is referred to as the ''source term'' and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. The accident source term is used to evaluate the radiological consequences of design basis accidents (DBAs) in showing compliance with various requirements of the NRC's regulations. Although originally used for site suitability analyses, the accident source term is a design parameter for accident mitigation features, equipment qualification, control room operator radiation doses, and post-accident vital area access doses. The measurement range and alarm setpoints of some installed plant instrumentation and the actuation of some plant safety features are based in part on the accident source term. The TID–14844 source term was explicitly stated as a required design parameter for several Three Mile Island (TMI)-related requirements.

The NRC's methods for calculating accident doses, as described in Regulatory Guide 1.3, ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors''; Regulatory Guide 1.4, ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors''; and NUREG–0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,'' were developed to be consistent with the TID–14844 source term and the whole body and thyroid dose guidelines stated in § 100.11. In this regulatory framework, the source term is assumed to be released immediately to the containment at the start of the postulated accident. The chemical form of the radioiodine released to the containment atmosphere is assumed to be predominantly elemental, with the remainder being small fractions of particulate and organic iodine forms. Radiation doses are calculated at the exclusion area boundary (EAB) for the first 2-hours and at the low population zone (LPZ) for the assumed 30-day duration of the accident. The whole body dose comes primarily from the noble gases in the source term. The thyroid dose is based on inhalation of radioiodines. In analyses performed to date, the thyroid dose has generally been limiting. The design of some engineered safety features, such as containment spray systems and the charcoal filters in the containment, the building exhaust, and the control room ventilation systems, are predicated on these postulated thyroid doses. Subsequently, the NRC adopted the whole body and thyroid dose criteria in Criterion 19 of 10 CFR Part 50, Appendix A (36 FR 3255; February 20, 1971).

The source term in TID–14844 is representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, NRC experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. Some of these additional accident analyses may involve source terms that are a fraction of those specified in TID–14844. The DBAs were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of the plant engineered safety features. These accident analyses are intentionally conservative in order to address known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired defense in depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods (60 FR 42622; August 16, 1995) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy.

Since the publication of TID–14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island (TMI). In 1995, the NRC published NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' which utilized this research to provide more physically based estimates of the accident source term that could be applied to the design of future light-water power reactors. The NRC sponsored significant review efforts by peer reviewers, foreign research partners, industry groups, and the general public (request for public comment was published in 57 FR 33374).

The information in NUREG–1465 presents a representative accident source term (''revised source term'') for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These revised source terms are described in terms of radionuclide composition and magnitude, physical and chemical form, and timing of release. Where TID–14844 addressed three categories of radionuclides, the revised source terms categorize the accident release into eight groups on the basis of similarity in chemical behavior. Where TID–14844 assumed an immediate release of the activity, the revised source terms have five release phases that are postulated to occur over several hours, with the onset of major core damage occurring after 30 minutes. Where TID–14844 assumed radioiodine to be predominantly elemental, the revised source terms assume radioiodine to be predominantly cesium iodide (CsI), an aerosol that is more amenable to mitigation mechanisms.

For DBAs, the NUREG–1465 source terms are comparable to the TID–14844 source term with regard to the magnitude of the noble gas and radioiodine release fractions. However, the revised source terms offer a more representative description of the radionuclide composition and release timing. The NRC has determined (SECY–94–302, dated December 1994) that design basis analyses will address the first three release phases—coolant, gap, and in-vessel. The ex-vessel and late in-vessel phases are considered to be unduly conservative for design basis analysis purposes. These latter releases could only result from core damage accidents with vessel failure and core-concrete interactions. The estimated frequencies of such scenarios are low enough that they need not be considered for the purpose of meeting the requirements of § 100.11 or, as proposed herein, § 50.67.

The objective of NUREG–1465 was to define revised accident source terms for regulatory application for future light water reactors. The NRC's intent was to capture the major relevant insights available from severe accident research to provide, for regulatory purposes, a more realistic portrayal of the amount of the postulated accident source term. These source terms were derived from examining a set of severe accident sequences for light water reactors (LWRs) of current design. Because of general similarities in plant and core design parameters, these results are considered to be applicable to evolutionary and passive LWR designs. The revised source term has been used in evaluating the Westinghouse AP–600 standard design certification application. (A draft version of NUREG–1465 was used in evaluating Combustion Engineering's (CE's) System 80+ design.)

The NRC considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID–14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source terms. The NRC also concluded that some licensees may wish to use an alternative source term in analyses to support operational flexibility and cost-beneficial licensing actions. The NRC initiated several actions to provide a regulatory basis for operating reactors to voluntarily amend their facility design bases to enable use of the revised source term in design basis analyses. First, the NRC solicited ideas on how an alternative source term might be implemented. In November 1995, the Nuclear Energy Institute (NEI) submitted its generic framework, Electric Power Research Institute Technical Report TR–105909, ''Generic Framework for Application of Revised Accident Source Term to Operating Plants.'' This report and the NRC response were discussed in SECY–96–242 (November 1996). Second, the NRC initiated a comprehensive assessment of the overall impact of substituting the NUREG–1465 source terms for the traditionally used TID–14844 source term at three typical facilities. This was done to evaluate the issues involved with applying the revised source terms at operating plants. SECY 98–154 (June 1998) described the conclusions of this assessment. Third, the NRC accepted license amendment requests related to implementation of the revised source terms at a small number of pilot plants. Experience has demonstrated that evaluation of a limited number of plant-specific submittals improves regulation and regulatory guidance development. The review of these pilot projects is currently in progress. Insights from these pilot plant reviews will be incorporated into the regulatory guidance that will be developed in conjunction with this rulemaking. Fourth, the NRC initiated an assessment on whether rulemaking would be necessary to allow operating reactors to use an alternative source term. The proposed rule and the supporting regulatory guidance that will be developed as part of this rulemaking have resulted from this assessment. The NRC plans to issue the supporting regulatory guidance for public comment on the same day as it publishes the final rule.

This proposed rulemaking for use of alternative source terms is applicable only to those facilities for which a construction permit was issued before January 10, 1997, under 10 CFR Part 50, ''Domestic Licensing of Production and Utilization Facilities.'' The regulations of this part are supplemented by those in other parts of Chapter I of Title 10, including Part 100, ''Reactor Site Criteria.'' Part 100 contains language that qualitatively defines a required accident source term and contains a note that discusses the availability of TID–14844. With the exception of § 50.34(f), there are no explicit requirements in Chapter I of Title 10 to use the TID–14844 accident source term. Section 50.34(f), which addresses additional TMI-related requirements, is only applicable to a limited number of construction permit applications pending on February 16, 1982, and to applications under Part 52.

An applicant for an operating license is required by § 50.34(b) to submit a final safety analysis report (FSAR) that describes the facility and its design bases and limits, and presents a safety analysis of the structures, systems, and components of the facility as a whole. Guidance in performing these analyses is given in regulatory guides. In its review of the more recent applications for operating licenses, the NRC has used the review procedures in NUREG-0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants'' (SRP). These review procedures reference or provide acceptable assumptions and analysis methods. The facility FSAR documents the assumptions and methods actually used by the applicant in the required safety analyses. The NRC's finding that a license may be issued is based on the review of the FSAR, as documented in the Commission's safety evaluation report (SER). By their inclusion in the FSAR, the assumptions (including the source term) become part of the design basis [1] of the facility. From a regulatory standpoint, the requirement to use the TID–14844 source term is expressed as a licensee commitment (typically to Regulatory Guide 1.3 or 1.4) documented in the facility FSAR, and is subject to the requirements of § 50.59.

In January 1997 (61 FR 65157), the NRC amended its regulations in 10 CFR Parts 21, 50, 52, 54, and 100. That regulatory action produced site criteria for future sites; presented a stable regulatory basis for seismic and geologic siting and the engineering design of future nuclear power plants to withstand seismic events; and relocated source term and dose requirements for future plants into part 50. Because these dose requirements tend to affect reactor design rather than siting, they are more appropriately located in Part 50. This decoupling of siting from design is consistent with the future licensing of facilities using standardized plan designs, the design features of which will be certified in a separate design certification rulemaking. This decoupling of siting from design was directed by Congress in the 1980 Authorization Act for the NRC. Because the revised criteria would not apply to operating reactors, the non-seismic and seismic reactor site criteria for operating reactors were retained as Subpart A and Appendix A to Part 100, respectively. The revised reactor site criteria were added as Subpart B in Part 100, and revised source term and dose requirements were moved to § 50.34. The existing source term and dose requirements of Subpart A of Part 100 will remain in place as the licensing bases for those operating reactors that do not elect to use an alternative source term.

In relocating the source term and dose requirements for future reactors to § 50.34, the NRC retained the requirements for the exclusion area and the low population zone, but revised the associated numerical dose criteria to replace the two different doses for the whole body and the thyroid gland with a single, total effective dose equivalent (TEDE) value. The dose criteria for the whole body and the thyroid, and the immediate 2-hour exposure period were largely predicated by the assumed source term being predominantly noble gases and radioiodines instantaneously released to the containment and the assumed ''single critical organ'' method of modeling the internal dose used at the time that Part 100 was originally published. However, the current dose criteria, by focusing on doses to the thyroid and the whole body, assume that the major contributor to doses will be radioiodine. Although this may be appropriate with the TID–14844 source term, as implemented by Regulatory Guides 1.3 and 1.4, it may not be true for a source term based on a more complete understanding of accident sequences and phenomenology.

The postulated chemical and physical form of radioiodine in the revised source terms is more amenable to mitigation and, as such, radioiodine may not always be the predominant radionuclide in an accident release. The revised source terms include a larger number of radionuclides than did the TID–14844 source term as implemented in regulatory guidance. The whole body and thyroid dose criteria ignore these contributors to dose. The NRC amended its radiation protection standards in Part 20 in 1991 (56 FR 23391; May 21, 1991) replacing the single, critical organ concept for assessing internal exposure with the TEDE concept that assesses the impact of all relevant nuclides upon all body organs. TEDE is defined to be the deep dose equivalent (for external exposure) plus the committed effective dose equivalent (for internal exposure). The deep dose equivalent (DDE) is comparable to the present whole body dose; the committed effective dose equivalent (CEDE) is the sum of the products of doses (integrated over a 50-year period) to selected body organs resulting from the intake of radioactive material multiplied by weighting factors for each organ that are representative of the radiation risk associated with the particular organ.

The TEDE, using a risk-consistent methodology, assesses the impact of all relevant nuclides upon all body organs. Although it is expected that in many cases the thyroid could still be the limiting organ and radioiodine the limiting radionuclide, this conclusion cannot be assured in all potential cases. The revised source terms postulate that the core inventory is released in a

[1] As defined in 10 CFR Part 50.2, 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. The NRC considers the accident source term to be an integral part of the design basis because it sets forth specific values (or range of values) for controlling parameters that constitute reference bounds for design.

sequence of phases over 10 hours, with the more significant release commencing at about 30 minutes from the start of the event. The assumption that the 2-hour exposure period starts immediately at the onset of the release is inconsistent with the phased release postulated in the revised source terms. The proposed rule would extend the future LWR dose criteria to operating reactors that elect to use an alternative source term.

An accidental release of radioactivity can result in radiation exposure to control room operators. Normal ventilation systems may draw this activity into the control room where it can result in external and internal exposures. Control room designs differ but, in general, design features are provided to detect the accident or the activity and isolate the normal ventilation intake. Emergency ventilation systems are activated to minimize infiltration of contaminated air and to remove activity that has entered the control room. Personnel exposures can also result from radioactivity outside of the control room. However, because of concrete shielding of the control room, these latter exposures are generally not limiting. The objective of the control room design is to provide a location from which actions can be taken to operate the plant under normal conditions and to maintain it in a safe condition under accident conditions. General Design Criterion 19 (GDC–19), ''Control Room,'' of Appendix A to 10 CFR part 50 (36 FR 3255; February 20, 1971), establishes minimum requirements for the design of the control room, including a requirement for radiation protection features adequate to permit access to and occupancy of the control room under accident conditions. The GDC–19 criteria were established for judging the acceptability of the control room design for protecting control room operators under postulated design basis accidents, a significant concern being the potential increases in offsite doses that might result from the inability of control room personnel to adequately respond to the event.

The GDC–19 criteria are expressed in terms of whole body dose, or its equivalent to any organ. The NRC did not revise the criteria when Part 20 was amended (56 FR 23391) instead deferring such action to individual facility licensing actions (NUREG/CR–6204). This position was taken in the interest of maintaining the licensing basis for those facilities already licensed. The NRC is proposing to replace the current GDC–19 dose criteria for future reactors and for operating reactors that elect to use an alternative source term with a criterion expressed in terms of TEDE. The rationale for this revision is similar to the rationale, discussed earlier in this preamble, for revising the dose criteria for offsite exposures.

On January 10, 1997 (61 FR 65157), the NRC amended 10 CFR Parts 21, 50, 52, 54, and 100 of its regulations to update the criteria used in decisions regarding power reactor siting for future nuclear power plants. The NRC intended that future licensing applications in accordance with Part 52 utilize a source term consistent with the source term information in NUREG–1465 and the accident TEDE criteria in Parts 50 and 100. However, during the final design approval (FDA) and design certification proceeding for the Westinghouse AP–600 advanced light-water reactor design, the NRC staff and Westinghouse determined that exemptions were necessary from §§ 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) and 10 CFR Part 50, Appendix A, GDC–19. This rule would eliminate the need for these exemptions for future applicants under Part 52 by making conforming changes to Part 50, Appendix A, GDC–19 and § 50.34.

## II. Objectives

The objectives of this proposed regulatory action are to—

1. Provide a regulatory framework for the voluntary implementation of alternative source terms as a change to the design basis at currently licensed power reactors, thereby enabling potential cost-beneficial licensing actions while continuing to maintain existing safety margins and defense in depth.
2. Retain the existing regulatory framework for currently licensed power reactor licensees who choose not to implement an alternative source term, but continue to comply with their existing source term.
3. Relocate source term and dose requirements that apply primarily to plant design into 10 CFR Part 50 for operating reactors that choose to implement an alternative source term, and
4. Implement conforming changes to § 50.34(f) and Part 50, Appendix A, GDC–19 to eliminate the need for exemptions for future applicants under Part 52.

## III. Alternatives

The first alternative considered by the NRC was to continue using current regulations for accident dose criteria and control room dose criteria. This is not considered to be an acceptable alternative. As discussed in the statements of consideration for the final siting rule (61 FR 65157, 65159; December 11, 1996), the NRC determined that dose criteria expressed in terms of whole body and thyroid doses were inconsistent with the use of new source terms not based upon TID–14844. With regard to the exclusion area dose guideline, the NRC had previously determined (id. at 65160) that the dose criterion applies to the 2-hour period resulting in the maximum dose.

The second alternative considered by the NRC was the replacement of the existing guidelines in § 100.11 and the existing criteria in 10 CFR Part 50 Appendix A, GDC–19 with revised dose criteria. This is not considered to be a desirable alternative because the provisions of the existing regulations form part of the licensing bases for many of the operating reactors. Therefore, these provisions must remain in effect for operating reactors that do not implement an alternative source term. In addition, this alternative would also be inconsistent with the NRC's philosophy of separating plant siting criteria and dose requirements.

The approach of establishing the requirements for use of alternative source terms in a new section to Part 50 while retaining the existing regulations in Part 100 Subpart A and Part 50 Appendix A GDC–19 was chosen as the best alternative.

The NRC considered alternatives with regard to providing regulatory guidance to support the new section to Part 50. The first option was to issue no additional regulatory guidance. This option was not considered to be acceptable because in the absence of clear regulatory guidance, licensee efforts in preparing applications and the NRC staff review of submitted applications, could be hindered by differences in interpretations and technical positions. This could result in the inefficient use of licensee and NRC staff resources, could cause licensing delays, and lead to less uniform and less consistent regulatory implementation.

The second option was to replace the existing regulatory guides that address the radiological consequences of accidents with new revisions. This is not considered to be an acceptable choice because the provisions of the existing regulatory guides form part of the licensing bases for many of the operating reactors. Therefore, these provisions must remain in effect for those operating reactors that do not implement an alternative source term. The third option was to issue a new regulatory guide on the implementation of alternative source terms that would include revised assumptions and acceptable analysis methods for each design basis accident in a series of appendices. The approach of issuing a new regulatory guide was determined to be the best option. To provide review guidance for the NRC staff, a new section on design basis radiological analyses using alternative source terms would be added to the Standard Review Plan.

## IV. Section-by-Section Analysis

### *A. Section 50.2*

The general ''definitions'' section for Part 50 would be supplemented by adding a definition of source term for the purpose of § 50.67. In NUREG–1465, the *source term* is defined by five projected characteristics: (1) Magnitude of radioactivity release, (2) radionuclides released, (3) physical form of the radionuclides released, (4) chemical form of the radionuclides released, and (5) timing of the radioactivity release. Although all five characteristics should be addressed in applications proposing the use of an alternative source term, there may be technically justifiable applications in which all five characteristics need not be addressed. The NRC intends to allow licensees flexibility in implementing alternative source terms consistent with maintaining a conservative, clear, logical, and consistent plant design basis. The regulatory guide that supports this proposed rule will contain guidance on an acceptable basis for defining the characteristics of an alternative source term.

### *B. Section 50.67(a)*

This paragraph would define the licensees that may seek to revise their current radiological source term with an alternative source term. The proposed rule is applicable only to holders of nuclear power plant operating licenses that were issued under 10 CFR Part 50 before January 10, 1997. The proposed rule would not require licensees to revise their current source term. The NRC considered the acceptability of the TID–14844 source term at current operating reactors and determined that the analytical approach based on the TID–14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach should not be required to reanalyze design basis accidents using a new source term. The proposed rule does not explicitly define an alternative source term. In lieu of an explicit reference to NUREG–1465, Footnote 1 to the proposed rule identifies the significant characteristics of an accident source term. The regulatory guide that will be issued to support this proposed rule will identify the NUREG–1465 source terms as acceptable alternatives to the source term in TID–14844, and will provide implementation guidance. This approach would provide for future revised source terms if they are developed and would allow licensees to propose additional alternatives for NRC consideration.

### *C. Section 50.67(b)(1)*

This paragraph of § 50.67 would state the information that a licensee must submit as part of a license amendment application to use an alternative source term. Because of the extensive use of the accident source term in the design and operation of a power reactor and the potential impact on postulated accident consequences and margins of safety of a change of such a fundamental design assumption, the NRC has determined that any change to the design basis to use an alternative source term should be reviewed and approved by the NRC in the form of a license amendment. Changes to the source term, by itself, would ordinarily constitute a no significant hazards consideration. In addition, generic analyses performed by the NRC staff in support of this proposed rule have indicated that there are potential changes to the facility as documented in the FSAR which would constitute a no significant hazards consideration. However, such determinations would have to be made for each proposed change based upon facility-specific evaluations. The procedural requirements for processing a license amendment are given in §§ 50.90 through 50.92.

The NRC's regulations provide a regulatory mechanism for a licensee to effect a change in its design basis in § 50.59. That section allows a licensee to make changes to the facility as described in the final safety evaluation report (FSAR) without prior NRC approval, unless the proposed change is deemed to involve an unreviewed safety question (USQ), or involves a change to the technical specifications incorporated into the facility license. If a USQ is determined to exist or if a change to the technical specifications is involved, the licensee must request NRC approval of the change using the license amendment process detailed in § 50.90. The criteria for determining that a USQ is involved appear in § 50.59. Significant to this proposed rule is the criterion that a USQ would exist if the proposed change resulted in an increase in consequences of an accident or malfunction. In many applications, alternative source terms may reduce the postulated consequences of the accident or malfunction. For this reason, the NRC determined that the regulatory framework of § 50.59 does not provide assurance that this change in the design basis would be recognized by the licensee as needing review by the NRC staff. After a licensee has been authorized to substitute an alternative source term in its design basis, subsequent changes to the facility that involve an alternative source term may be processed under § 50.59 or § 50.90, as appropriate. However, a subsequent change to the source term itself could not be implemented under § 50.59; in all cases a change to the source term must be made through a license amendment.

The proposed rule would require the applicant to perform analyses of the consequences of applicable design basis accidents previously analyzed in the safety analysis report and to submit a description of the analysis inputs, assumptions, methodology, and results of these analyses for NRC review. Applicable evaluations may include, but are not limited to, those previously performed to show compliance with § 100.11, § 50.49, Part 50 Appendix A GDC–19, § 50.34(f), and NUREG–0737 requirements II.B.2, II.B.3, III.D.3.4. The regulatory guide that supports this proposed rule will provide guidance on the scope and extent of analyses used to show compliance with this rule and on the assumptions and methods used therein. It is not the NRC's intent that all of the design basis radiological analyses for a facility be performed again as a prerequisite for approval of the use of an alternative source term. The NRC does expect that the applicant will perform sufficient evaluations, supported by calculations as warranted, to demonstrate the acceptability of the proposed amendment.

### *D. Sections 50.67(b)(2)(i), (ii), (iii)*

These subparagraphs would contain the three criteria for NRC approval of the license amendment to use an alternative source term. A detailed rationale for the use of 0.25 Sv (25 rem) TEDE as an accident dose criterion and the use of the 2-hour exposure period resulting in the maximum dose for future LWRs is provided at 61 FR 65157; December 11, 1996. The same considerations that formed the basis for that rationale are similarly applicable to operating reactors that elect to use an alternative source term. The NRC believes that it is technically appropriate and logical to extend the philosophy of decoupling of design and siting, and the dose criteria established for future LWRs to operating reactors that elect to use an alternative source term.

The NRC is proposing to replace the current GDC–19 dose criteria for operating reactors that elect to use an alternative source term with a criterion of 0.05 Sv (5 rem) TEDE for the duration of the accident. This criterion would be included in § 50.67 rather than GDC–19 in order to co-locate all of the dose requirements associated with alternative source terms. The bases for the NRC's decision are: first, that the criteria in GDC–19 and that in the proposed rule are based on a primary occupational exposure limit. Second, the language in GDC–19: ''5 rem whole body, or its equivalent to any part of the body'' is subsumed by the definition of TEDE in § 20.1003 and by the 0.05 Sv (5 rem) TEDE annual limit in § 20.1201(a). Although the weighting factors stated in § 20.1003 for use in determining TEDE differ in magnitude from the weighting factors implied in the 0.3 Sv (30 rem) thyroid criteria used for showing compliance with GDC–19, these differences are the result of improvement in the science of assessing internal exposures and do not represent a reduction in the level of protection. Third, as discussed earlier, the use of TEDE in conjunction with alternative source terms has been deemed appropriate and necessary. Fourth, the use of TEDE for the control room dose criterion is consistent with the use of TEDE in the accident dose criteria for offsite exposure.

The NRC is not including a ''capping'' limitation, an additional requirement that the dose to any individual organ not be in excess of some fraction of the total as provided for routine occupational exposures. The bases for the NRC's decision are: first, that this non-inclusion of a ''capping'' limitation is consistent with the final rule published in December 11, 1996 (61 FR 65157), with regard to doses to persons offsite. Second, the use of 0.05 Sv (5 rem) TEDE as the control room criterion does not imply that this would be an acceptable exposure during emergency conditions, or that other radiation protection standards of Part 20, including individual organ dose limits, might not apply. This criterion is provided only to assess the acceptability of design provisions for protecting control room operators under postulated DBA conditions. The DBA conditions assumed in these analyses, although credible, generally do not represent actual accident sequences but are specified as conservative surrogates to create bounding conditions for assessing the acceptability of engineered safety features. Third, § 20.1206 permits a once-in-a-lifetime planned special dose of five times the annual dose limits. Also, Environmental Protection Agency (EPA) guidance sets a limit of five times the annual dose limits for workers performing emergency services such as lifesaving or protection of large populations. Considering the individual organ weighting factors of § 20.1003 and assuming that only the exposure from a single organ contributed to TEDE, the organ dose, although exceeding the dose specified in § 20.1201(a), would be less than that considered acceptable as a planned special dose or as an emergency worker dose. The NRC is not suggesting that control room dose during an accident can be treated as a planned special exposure or that the EPA emergency worker dose limits are an alternative to GDC–19 or the proposed rule. However, the NRC does believe that these provisions offer a useful perspective that supports the conclusion that the organ doses implied by the proposed 0.05 Sv (5 rem) criterion can be considered to be acceptable due to the relatively low probability of the events that could result in doses of this magnitude.

Although the dose criteria in the proposed rule would supersede the dose criteria in GDC–19, the other provisions of GDC–19 remain applicable.

*E. 10 CFR Part 50, Appendix A, GDC–19*

GDC–19 would be changed to include the TEDE dose criterion for control room design for applicants for construction permits, design certifications, and combined operating licenses that submitted applications after January 10, 1997 (the effective date of the 1996 rulemaking adopting the TEDE criterion), and for those licenses using an alternative source term under § 50.67. The proposed change to GDC–19 addresses the use of alternative source terms at operating reactors and a deficiency identified in the regulatory framework for early site permits, standard design certifications, and combined licenses under part 52. Sections 52.18, 52.48, and 52.81 establish that applications filed under part 52, Subparts A, B, and C, respectively, will be reviewed according to the standards given in 10 CFR parts 20, 50, 51, 55, 73, and 100 to the extent that those standards are technically relevant to the proposed design. Therefore, GDC–19 is pertinent to applications under part 52. The final rule that became effective on January 10, 1997 (61 FR 65157; December 11, 1996), established accident TEDE criteria (in § 50.34) for applicants under part 52 but did not change the existing control room whole body (or equivalent) dose criterion in GDC–19. Thus, exemptions from the dose criteria in the current GDC–19 were necessary in the design certification process for the Westinghouse AP–600 advanced LWR in order to use the 0.05 Sv (5 rem) TEDE criterion deemed necessary for use with alternative source terms. Exemptions would arguably be necessary for future applicants for construction permits, design certifications, and combined operating licenses. This proposed change would eliminate the need for these exemptions.

*F. Sections 21.3, 50.2, 50.49(b)(1)(i)(C), 50.65(b)(1), and 54.4(a)(1)(iii)*

These sections would be revised to conform with the relocation of accident dose criteria from § 100.11 to § 50.67 for operating reactors that have amended their design bases to use an alternative source term.

*G. Section 50.34*

A new footnote to § 50.34 would be added to define what constitutes an accident source term. This new footnote is identical to the existing footnote 1 to § 100.11, and is being added to provide for consistency between Parts 50 and 100.

*H. Sections 50.34(f)(2)(vii), (viii), (xxvi) and (xxviii)*

These paragraphs would be revised to replace an explicit reference to the ''TID–14844 source term'' with a more general reference to ''accident source term.'' These changes potentially affect two classes of applicants. The first affected class is facilities that obtain combined licenses under part 52. Section 52.47(a)(ii) states that applications for combined licenses must contain, inter alia, ''demonstration of compliance with any technically-relevant portions of the Three Mile Island requirements set forth in § 50.34(f).'' Section 50.34(f) contains several references to the TID–14844 source term. These references would be modified to delete the reference to TID–14844. This would make it clear that applicants for combined licenses would not use the TID–14844 source term but would use the source term in the referenced design certification, or a source term that is justified in the combined license application.

The second affected class is the small subset of plants that had construction permits pending on February 16, 1982. With the proposed change, these plants could use either the TID–14844 source term or an alternative source term in their operating license applications.

## V. Future Regulatory Action

The NRC is developing the following regulatory guides and Standard Review Plan sections to provide prospective applicants with the necessary guidance for implementing the proposed regulation. The draft guide and draft Standard Review Plan section will be issued to coincide with the publication of the final regulations that would implement this proposed rulemaking. A notice of availability for these materials will be published in the **Federal Register** at a future date.

*1. Draft Guide DG–1081, ''Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors''*

This guide is expected to present regulatory guidance on the implementation of an alternative source term at an operating reactor. The guide is expected to address issues involving limited or selective implementation of an alternative source term and probabilistic risk assessment (PRA) issues related to plant modifications based on an alternative source term, and to provide guidance on the scope and extent of affected DBA radiological analyses and associated acceptance criteria. The guide is expected to include revised assumptions and methods for each affected DBA in a series of appendices. These appendices will supersede the guidance in Regulatory Guides 1.3, 1.4, 1.25, and 1.77, and will supplement guidance in Regulatory Guide 1.89 for those facilities using an alternative source term.

*2. Standard Review Plan Section, 15.0.1, ''Radiological Consequence Analyses Using Alternative Source Terms''*

This SRP section presents guidance to NRC staff in the review of the adequacy of licensee submittals requesting approval for use of an alternative source term.

## VI. Referenced Documents

Copies of NUREG–0737, NUREG–0800, NUREG–1465, and NUREG/CR–6204 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Mail Stop SSOP, Washington, DC 20402–9328. Copies also are available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy also is available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street, NW (Lower Level), Washington, DC.

Copies of issued regulatory guides may be purchased from the Government Printing Office (GPO) at the current GPO price. Information on current GPO prices may be obtained by contacting the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402–9328. Issued guides also may be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by writing NTIS, 5826 Port Royal Road, Springfield, VA 22161.

Copies of SECY–94–302, SECY–96–242, SECY–98–154, TID14844, and TR–105909 are available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street, NW (Lower Level), Washington, DC.

## VII. Draft Finding of No Significant Environmental Impact: Availability

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the NRC's regulations in Subpart A of 10 CFR Part 51, that this regulation is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. This proposed rule would allow operating reactors to replace the traditional TID–14844 source term with a more realistic source term based on the insights gained from extensive accident research activities. The actual accident sequence and progression would not be changed; it is the regulatory assumptions regarding the accident that would be affected by the change. The use of an alternative source term alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF) or actual offsite or onsite radiation doses. An alternative source term could be used to justify changes in the plant design that might have an impact on CDF or LERF or that might increase offsite or onsite doses. These potential changes are subject to existing requirements in the NRC's regulations. Thus, the level of protection of public health and safety provided in NRC regulations would not be decreased by this proposed rule. The proposed rule would not affect non-radiological plant effluents and would have no significant environmental impact.

As discussed above, the determination of the environmental assessment is that there would be no significant offsite impact on the public from this action. However, the general public should note that the NRC welcomes public participation. Also, the NRC has committed itself to complying in all its actions with Executive Order (E.O.) 12898, ''Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations,'' dated February 11, 1994. In accordance with that Executive Order, the NRC has determined that there are no disproportionately high and adverse impacts on minority and low income parties. In the letter and spirit of E.O. 12898, the NRC is requesting public comments on any environmental justice considerations or questions that the public thinks may be related to this proposed rule, but that somehow were not addressed. The NRC uses the following working definition of environmental justice: Environmental justice means the fair treatment and meaningful involvement of all people, regardless of race, ethnicity, culture, income, or educational level with respect to the development, implementation and enforcement of environmental laws, regulations, and policies. Comments on any aspect of the environmental assessment, including environmental justice, may be submitted to the NRC as indicated under the **ADDRESSES** heading.

The draft environmental assessment and the draft finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are available from Mr. Stephen F. LaVie, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory NRC, Washington, DC 20555–0001, telephone: 301–415–1081, or by Internet electronic mail to sfl@nrc.gov.

## VIII. Paperwork Reduction Act Statement

This proposed rule increases the burden on licensees by requiring that when seeking to revise their current accident source term in design basis radiological consequence analyses, they apply for an amendment under § 50.90. The public burden for this information collection is estimated to average 609 hours per request. Because the burden for this information collection is insignificant, Office of Management and Budget (OMB) clearance is not required. Existing requirements were approved by the Office of Management and Budget, approval number 3150-0011.

*Public Protection Notification*

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

## IX. Regulatory Analysis

The Commission has prepared a regulatory analysis on this regulation. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis are available from Mr. Stephen F. LaVie, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–1081, or by Internet electronic mail to sfl@nrc.gov.

## X. Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this regulation will not have a significant economic impact on a substantial number of small entities. This proposed regulation will affect only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the definition of ''small entities'' found in the Regulatory Flexibility Act or within the size standards established by the NRC (April 11, 1995; 60 FR 18344).

## XI. Backfit Analysis

The NRC has determined that the backfit rule in 10 CFR 50.109, does not apply to this proposed regulation and that a backfit analysis is not required for this proposed regulation because these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1). This proposed regulation amends the NRC's regulations by establishing alternate requirements that may be voluntarily adopted by licensees.

## List of Subjects

*10 CFR Part 21*

Nuclear power plants and reactors, Penalties, Radiation protection, Reporting and recordkeeping requirements.

*10 CFR Part 50*

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

*10 CFR Part 54*

Administrative practice and procedure, Age-related degradation, Backfitting, Classified information, Criminal penalties, Environmental protection, Nuclear power plants and reactors, Reporting and recordkeeping requirements.

For the reasons noted in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 553, the NRC is proposing the following amendments to 10 CFR Parts 21, 50, and 54:

## PART 21—REPORTING OF DEFECTS AND NONCOMPLIANCE

1. The authority citation for part 21 continues to read as follows:

**Authority:** Sec. 161, 68 Stat. 948, as amended, sec. 234, 83 Stat. 444, as amended, sec. 1701, 106 Stat. 2951, 2953 (42 U.S.C. 2201, 2282, 2297f); secs. 201, as amended, 206, 88 Stat. 1242, as amended, 1246 (42 U.S.C. 5841, 5846).

Section 21.2 also issued under secs. 135, 141, Pub. L. 97–425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161).

2. Section 21.3 is amended by republishing the introductory text and revising paragraph (1)(i)(C) of the definition of *Basic component* to read as follows:

### § 21.3 Definitions.

As used in this part:

*Basic component.* (1)(i) * * *

(C) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

* * * * *

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

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

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

Section 50.7 also issued under Pub. L. 95–9601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91–9190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91–9190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–9415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C 2237).

4. Section 50.2 is amended by republishing the introductory text, by revising paragraph (1)(iii) of the definition of *Basic component* and by adding in alphabetical order the definition for *Source term* to read as follows:

### § 50.2 Definitions.

As used in this part,

* * * * *

*Basic component* * * *

(1) * * *

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

* * * * *

*Source term* refers to the magnitude and mix of radionuclides released from the reactor core to the reactor containment, their physical and chemical form, and the timing of their release.

* * * * *

5. Section 50.34 is amended by revising paragraphs (f)(2)(vii), (viii), (xxvi), and (xxviii) to read as follows:

### § 50.34 Contents of applications; technical information.

* * * * *

(f) * * *

(2) * * *

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term [11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term [12] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to

[11] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[12] See footnote 11 to paragraph (f)(2)(vii) of this section.

the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

* * * * *

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term [13] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

* * * * *

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term [14] release, and make necessary design provisions to preclude such problems. (III.D.3.4)

6. Section 50.49 is amended by revising paragraph (b)(1)(i)(C) to read as follows:

### § 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

* * * * *

(b) * * *

(1) * * *

(i) * * *

(C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

* * * * *

7. Section 50.65 is amended by revising paragraph (b)(1) to read as follows:

### § 50.65 Requirements for monitoring the effectiveness of maintenance at nuclear power plants.

* * * * *

(b) * * *

(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

* * * * *

[13] See footnote 11 to paragraph (f)(2)(vii) of this section.

[14] See footnote 11 to paragraph (f)(2)(vii) of this section.

8. Part 50 is amended by adding § 50.67 to read as follows:

### § 50.67 Accident source term.

(a) *Applicability.* The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) *Requirements.* (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents [1] previously analyzed in the safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) [2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

9. Part 50, Appendix A, II., General Design Criterion 19, is revised to read as follows:

### Appendix A to Part 50—General Design Criteria for Nuclear Power Plants

* * * * *

II. * * *

*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.

Applicants for construction permits under this part or a design certification or combined license under part 52 of this chapter who apply on or after January 10, 1997, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

* * * * *

## PART 54—REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR NUCLEAR POWER PLANTS

10. The authority citation for part 54 continues to read as follows:

**Authority:** Secs. 102, 103, 104, 161, 181, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs 201, 202, 206, 88 Stat. 1242, 1244, as amended (42 U.S.C. 5841, 5842), E.O. 12829, 3 CFR, 1993 Comp., p. 570; E.O. 12958, as amended, 3 CFR, 1995 Comp., p. 333; E.O. 12968, 3 CFR, 1995 Comp., p. 391.

11. Section 54.4 is amended by revising paragraph (a)(1)(iii) to read as follows:

**§ 54.4 Scope.**

(a) * * *

(1) * * *

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

* * * * *

Dated at Rockville, Maryland, this 5th day of March 1999.

For the Nuclear Regulatory Commission.

**Annette Vietti-Cook,**

*Secretary of the Commission.*

[FR Doc. 99–6058 Filed 3–10–99; 8:45 am]

BILLING CODE 7590–01–U

## DEPARTMENT OF TRANSPORTATION

## Federal Aviation Administration

## 14 CFR Part 71

**[Airspace Docket No. 99–ANM–02]**

## Proposed Revision of Class E Airspace; Colstrip, MT

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

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

**SUMMARY:** This proposal would amend the Colstrip, MT, Class E area and provide additional controlled airspace to accommodate the development of new Standard Instrument Approach Procedures (SIAP) utilizing the Global Positioning System (GPS) at the Colstrip, Airport.

**DATES:** Comments must be received on or before April 26, 1999.

**ADDRESSES:** Send comments on the proposal in triplicate to: Manager, Airspace Branch, ANM–520, Federal Aviation Administration, Docket No. 99–ANM–02, 1601 Lind Avenue SW, Renton, Washington, 98055–4056.

The official docket may be examined in the office of the Assistant Chief Counsel for the Northwest Mountain Region at the same address.

An informal docket may also be examined during normal business hours in the office of the Manager, Air Traffic Division, Airspace Branch, at the address listed above.

**FOR FURTHER INFORMATION CONTACT:**
Dennis Ripley, ANM–520.6, Federal Aviation Administration, Docket No. 99–ANM–02, 1601 Lind Avenue SW, Renton, Washington 98055–4056; telephone number: (425) 227–2527.

**SUPPLEMENTARY INFORMATION:**

### Comments Invited

Interested parties are invited to participate in this proposed rulemaking by submitting such written data, views, or arguments, as they may desire. Comments that provide the factual basis supporting the views and suggestions presented are particularly helpful in developing reasoned regulatory decisions on the proposal. Comments are specially invited on the overall regulatory, aeronautical, economic, environmental, and energy related aspects of the proposal. Communications should identify the airspace docket number and be submitted in triplicate to the address listed above. Commenters wishing the FAA to acknowledge receipt of their comments on this notice must submit, with those comments, a self-addressed stamped postcard on which the following statement is made: ''Comments to Airspace Docket No. 99–ANM–02.'' The postcard will be date/time stamped and returned to the commenter. All communications received on or before the specified closing date for comments will be considered before taking action on the proposed rule. The proposal contained in this notice may be changed in the light of comments received. All comments submitted will be available for examination at the address listed above both before and after the closing date for comments. A report summarizing each substantive public contact with FAA personnel concerned with this rulemaking will be filed in the docket.

### Availability of NPRM's

Any person may obtain a copy of this NPRM by submitting a request to the Federal Aviation Administration, Airspace Branch, ANM–520, 1601 Lind Avenue SW, Renton, Washington 98055–4056. Communications must identify the notice number of this NPRM. Persons interested in being placed in a mailing list for future NPRM's should also request a copy of Advisory Circular No. 11–2A, which describes the application procedure.

### The Proposal

The FAA is considering an amendment to Title 14 Code of Federal Regulations, part 71 (14 CFR part 71) by revising Class E airspace at Colstrip, MT, in order to accommodate two new GPS SIAP to the Colstrip Airport. This amendment would provide additional airspace by lowering the Class E area to the west in order to meet current criteria standards associated with SIAP holding patterns. The FAA establishes Class E airspace where necessary to contain aircraft transitioning between the terminal and en route environments. The intended effect of this proposal is designed to provide safe and efficient use of the navigable airspace and to promote safe flight operations under Instrument Flight Rules (IFR) at the Constrip Airport and between the terminal and en route transition stages.

The area would be depicted on aeronautical charts for pilot reference. The coordinates for this airspace docket are based on North American Datum 83. Class E airspace areas extending upward from 700 feet or more above the surface of the earth, are published Paragraph 6005, of FAA Order 7400.9F dated September 10, 1998, and effective September 16, 1998, which is incorporated by reference in 14 CFR 71.1. The Class E airspace designation listed in this document would be published subsequently in the Order.

The FAA has determined that this proposed regulation only involves an established body of technical regulations for which frequent and routine amendments are necessary to keep them operationally current. It, therefore, (1) is not a ''significant regulatory action'' under Executive Order 12866; (2) is not a ''significant rule'' under DOT Regulatory Policies and Procedures (44 FR 11034; February 26, 1979); and (3) does not warrant preparation of a Regulatory Evaluation as the anticipated impact is so minimal. Since this is a routine matter that will only affect air traffic procedures and air navigation, it is certified that this rule, when promulgated, will not have a significant economic impact on a substantial number of small entities under the criteria of the Regulatory Flexibility Act.

### List of Subjects in 14 CFR Part 71

Airspace, Incorporation by reference, Navigation (air).

### The Proposed Amendment

In consideration of the foregoing, the Federal Aviation Administration proposes to amend 14 CFR part 71 as follows:

## PART 71—DESIGNATION OF CLASS A, CLASS B, CLASS C, CLASS D, AND CLASS E AIRSPACE AREAS; AIRWAYS; ROUTES, AND REPORTING POINTS

1. The authority citation for 14 CFR part 71 continues to read as follows:

**Authority:** 49 U.S.C. 106(g), 40103, 40113, 40120; E.O. 10854, 24 FR 9565, 3 CFR, 1959–1963 Comp., p. 389.