any adverse comments, or expressions of intent to submit adverse comments, within the scope of the rulemaking, FSIS is affirming the November 30, 1999 effective date for this direct final rule.

EFFECTIVE DATES: November 30, 1999.

FOR FURTHER INFORMATION CONTACT: Daniel L. Engeljohn, Director, Regulations Development and Analysis Division, Office of Policy, Program Development, and Evaluation, Food Safety and Inspection Service, U.S. Department of Agriculture, (202) 720–5627.

SUPPLEMENTARY INFORMATION:

Background

On October 1, 1999, FSIS published a direct final rule, ''Scale Requirements for Accurate Weights, Repairs, Adjustments, and Replacement After Inspection'' (64 FR 53186). This direct final rule notified the public of FSIS's intention to amend the Federal meat and poultry products inspection regulations to update references to the National Institute of Standards and Technology (NIST) Handbook 44, ''Specifications, Tolerances, and Other Technical Requirements for Weighing and Measuring Devices.'' The 1999 edition of NIST Handbook 44 was published in November 1998 and is the most current edition of the handbook. FSIS is amending the provisions in its regulations that reference NIST Handbook 44 to reflect this most recent edition.

After publication of the direct final rule, the American Meat Institute (AMI), a national trade association representing packers and processors of meat and poultry products, contacted FSIS to express a minor concern associated with the rulemaking. AMI noted that Section 2.24 Automatic Weighing Systems of the 1999 edition of NIST Handbook 44 is a tentative code, has only a trial or experimental status, and is not intended to be enforced by weights and measures officials. AMI expressed concern that FSIS inspection program employees would not interpret Section 2.24 as tentative and would enforce the requirements of Section 2.24 against existing equipment in meat and poultry plants before it is adopted as a permanent code. AMI requested that, prior to the effective date of the rule, FSIS issue some kind of notification to its inspection program personnel explaining that Section 2.24 is a tentative code and is not enforceable against existing equipment.

The direct final rule updates a document, NIST Handbook 44, that has previously been approved for incorporation by reference in the Code of Federal Regulations. The current FSIS regulations reference the 1994 edition of NIST Handbook 44, published in November 1993. The 1994 edition of the handbook does not include Section 2.24 Automatic Weighing Systems. In the 1999 edition of Handbook 44, it is clearly stated that Section 2.24 has only a trial or experimental status, and that it is not intended to be enforced by weights and measures officials. However, Section 2.24 is intended to be used by the National Type Evaluation Program for type evaluation of automatic weighing systems, which permits these devices to be tested to ensure conformance with a nationally accepted standard.

When FSIS issues new regulations, it provides the new or revised regulations to inspection program employees. The Agency also provides inspection program employees with the necessary implementing instructions. Therefore, FSIS will issue notification to the field employees explaining that Section 2.24 Automatic Weighing Systems is a tentative code, and that it is not to be enforced until it is upgraded to become a permanent code.

Because FSIS did not receive any adverse comments or intent to submit adverse comments in response to the direct final rule, the effective date remains as November 30, 1999.

Done at Washington, DC, on: December 14, 1999.

Thomas J. Billy,

Administrator.

[FR Doc. 99–33205 Filed 12–22–99; 8:45 am]

BILLING CODE 3410–DM–P

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: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending 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 will 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 announcing the availability of a draft regulatory guide and a draft Standard Review Plan section on this subject for public comment. The NRC is also amending its regulations to revise certain sections to conform with the final rule published on December 11, 1996, concerning reactor site criteria.

EFFECTIVE DATE: January 24, 2000.

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 (or, for early reactors, a hazard summary 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. Changes to the design of the facility and the procedures for operating the facility are evaluated in part by determining whether there are changes to the calculated fission product release.

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 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; July 28, 1992).

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 (up to and including the early in-vessel phase) 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, December 19, 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 inappropriate for design basis analysis purposes. These latter releases could only result from core damage accidents with vessel failure and core-concrete interactions.

The objective of NUREG–1465 was to define revised accident source terms for regulatory application for future light water reactors (LWRs). 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 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 AP600 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 concluded that some licensees may wish to use an alternative source term in analyses to support operational flexibility and cost-beneficial licensing actions and that some of these applications could provide concomitant improvements in overall safety and in reduced occupational exposure. 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 25, 1996). Second, the NRC initiated an 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 30, 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 have been incorporated into the regulatory guidance that was 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. This final rule and the supporting regulatory guidance have resulted from this assessment.

This final rulemaking for use of alternative source terms is applicable to holders of operating licenses issued prior to January 10, 1997, under 10 CFR Part 50, ''Domestic Licensing of Production and Utilization Facilities,'' and to holders of renewed licenses under 10 CFR Part 54, ''Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'' whose initial operating license was issued prior to January 10, 1997. The regulations of Part 50 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). Fundamental assumptions that are design inputs, including the source term, were required to be included in the FSAR and became 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 1996 (61 FR 65175; December 11, 1996), 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 plant designs, the design features of which have been or will be certified in a separate design certification rulemakings. 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

[1] As defined in § 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.

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 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 final rule adopts the future LWR dose criteria for 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; May 21, 1991) instead deferring such action to individual facility licensing actions (NUREG/CR–6204, ''Questions and Answers Based on the Revised 10 CFR Part 20''). This position was taken in the interest of maintaining the licensing basis for those facilities already licensed. The NRC is replacing the current dose criteria of GDC–19 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 AP600 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 final 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. Analysis of Public Comments

The NRC published a proposed rule in the **Federal Register** (64 FR 12117, March 31, 1999); that would provide a regulatory framework for the voluntary implementation of alternative source terms as a change to the design basis at currently licensed power reactors, while retaining the existing regulatory framework for currently licensed power reactor licensees who choose not to implement an alternative source term. The rule proposed relocating 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. The rule also proposed 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.

The NRC received seven letters commenting on the proposed rule. All comments including those received by the NRC after the expiration of the public comment period but before June 25, 1999, were considered. The commenters included two State regulatory agencies, two nuclear industry groups and three utilities. The State of Florida Department of Community Affairs indicated that they had no comments on the proposed rule. The State of New Jersey Department of Environmental Protection concurred with the NRC's position on the use of an AST in emergency preparedness applications and stated a desire to review the draft regulatory guidance when issued. Winston & Strawn submitted comments on behalf of the Nuclear Utility Backfitting and Reform Group (NUBARG). The Nuclear Energy Institute (NEI) submitted comments on behalf of the nuclear industry. Two of the utilities provided comments, while the third endorsed the comments submitted by NEI. Copies of these letters are available for public inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

1. NUBARG Comments

NUBARG supports the rule, noting that the rule as proposed defines an acceptable regulatory process for implementing more realistic accident source terms. NUBARG requested clarification in the final rule of situations in which an alternative source term (AST) may be applied in future backfitting[2] decisions. First, NUBARG suggests that the NRC clarify the extent it intends to use the revised source term in assessing whether new generic requirements provide a cost-justified, substantial increase in safety in accordance with NRC's backfitting rule, § 50.109. NUBARG believes that continued use of the source term in TID–14844 for this purpose in spite of its known limitations would be inappropriate and could lead to overly conservative estimates of the safety impact of proposed new requirements. Second, NUBARG suggests a similar clarification for plant-specific backfit decisions for plants that have not opted to implement the revised source term. NUBARG believes that the NRC has discretion to take all relevant factors into account in making its safety benefit assessment of the proposed backfit, including the current state of knowledge concerning the accident source term. NUBARG suggested that the statements of considerations accompanying the final rule address these issues. NUBARG also suggests that relevant NRC guidance should also be revised to reflect NRC policy in these areas.

NRC Response. When radiological consequence analyses are involved, the NRC expects to use a technically appropriate AST in evaluating generic and plant-specific backfitting analyses, including those proposed for facilities that have not implemented an AST. The NRC agrees with the NUBARG position that the NRC has discretion to take all new information on accident source terms into account. The NRC's guidance for evaluating proposed NRC regulatory actions (including backfitting) are contained in NUREG/BR–0058, ''Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,'' and NUREG/BR–0184, ''Regulatory Analysis Technical Evaluation Handbook.'' These documents state that value and impact (including adverse effects on health and safety) parameters are to be best estimates, preferably mean or expected values. These documents also provide that analyses are to be based largely on risk considerations.

2. NEI Comment 1

NEI stated that the Section-by-Section Analysis in the proposed rule notice is consistent with the NRC's intent to permit limited application of the new research results. NEI noted that these limited applications are of two types: (1) application of alternative source term radiological composition and magnitude in a quantitative analysis relative to the effect on the performance of a given engineered safety feature; or (2) application of only the timing aspects in conjunction with the original TID–14844 source term. NEI stated that proposed § 50.67 appears to apply to applications where a licensee would use a completely new source term such as NUREG–1465 in all aspects of the plant design. The NEI comment acknowledged that further guidance in a subsequent regulatory guide and standard review plan is helpful and necessary. Nonetheless, NEI is concerned that licensee pursuit of either of these limited applications might ultimately require seeking an exemption to § 50.67, or require extensive analysis. NEI recommended that the NRC should: (1) revise the proposed rule language to accommodate limited application of an alternative source term as done in the Section-By-Section Analysis; (2) provide clarification in the Statement of Consideration (SOC) for the rule; and (3) for applications that continue to use the TID source term but incorporate attributes of newer technical insights such as timing of releases, specify that the provisions of the proposed rule do not apply.

NRC Response. The language of § 50.67(b) requires an evaluation of the consequences of applicable design basis accidents. The NRC believes that the use of the modifier applicable provides the basis for processing selective implementations. Design basis accidents not applicable to a particular selective implementation would not be required to be evaluated. The NRC expects that the licensee will evaluate all applicable impacts of the proposed AST implementation. While a selective implementation may result in a reduced scope of evaluation, the licensee must still demonstrate that the AST implementation and any associated proposed modifications will not result in accident conditions exceeding the criteria specified in § 50.67. Therefore, these criteria are applicable to full and selective implementations alike. The scope of the required re-analyses will depend on the specific application proposed by the licensee. Guidance with regard to this scope is properly provided in the draft regulatory guide prepared for this rule. Therefore, the NRC has decided against revising the rule language as suggested by NEI. Consistent with the second NEI recommendation, the NRC has modified paragraph D of the section-by-section analysis to clarify this issue.

3. NEI Comment 2

In its second comment, NEI noted that the SOC provides that licensees may need to perform additional evaluations of equipment qualifications (§ 50.49). The SOC should discuss the circumstances when such an evaluation may be necessary. NEI recommended that the SOC should be amended to state that regardless of source term used, the licensee would be required to re-evaluate the equipment qualification only when a plant modification alters the plant configuration so that the underlying assumptions, with respect to dose distribution and effects, are materially altered. NEI summarized conclusions of several references in support of its position. NEI stated that there is no basis to require or expect additional analyses of equipment qualification if a licensee applied the alternative source term in limited scope applications, absent a plant configuration change that materially alters the dose distribution and effects assumed in existing analyses.

NRC Response. The re-baselining study prepared by the NRC staff (SECY–98–154, June 30, 1998) considered the impact of an AST on analyses of the postulated integrated radiation doses for plant components exposed to containment atmosphere radiation sources and those exposed to containment sump radiation sources. The staff's conclusions regarding the atmosphere sources are consistent with those identified by NEI in its comment. However, the re-baselining study also concluded that the increased concentration of cesium in the containment sump water could result in

[2] As provided in § 50.109, Backfitting is defined as the modification of or addition to systems, structures, components, or the design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position.

an increase in the postulated integrated radiation doses for certain plant components subject to equipment qualification. It is because of this conclusion that the NRC included the discussion in the SOC regarding re-evaluation of equipment environmental qualification. The NEI comment provides no additional information that would cause the NRC to change its position on this matter. Further, the NRC has determined that it is necessary to consider the potential impact of the postulated cesium concentration in the containment sump water as it applies to all operating power reactors, not just to those licensees amending their design basis to use an AST. Since the postulated increase in the integrated dose occurs only following an accident, there is no adverse effect on equipment relied upon to perform safety functions immediately following an accident. Rather, this issue affects equipment that is required to be operable longer than about 30 days to 4 months after an accident. As such, the NRC determined that continued plant operation does not pose an immediate threat to public health and safety. Also, should such long-term equipment fail there will not be an undue threat to public health and safety as protective actions for the public would have already been implemented by the time the postulated failure could occur. In addition, the time period between the onset of the event and the projected failure allows compensatory measures to be taken to prevent the equipment failure or to restore the degraded safety function. The NRC will evaluate this issue as a generic safety issue to determine whether further regulatory actions are justified. The final regulatory guide, or subsequent revisions thereto, is expected to reflect the resolution of this generic safety issue.

4. NEI Comment 3

NEI recommends that the definition of Source Term in § 50.2 be revised to ''Source term refers to the magnitude and mix of radionuclides released from the fuel, their physical and chemical form, and the timing of their release.'' NEI stated that the language in the proposed rule would prohibit the use of § 50.67 for accidents such as the fuel handling accident.

NRC Response. The NRC agrees with the proposed revision. The proposed definition was consistent with the definition of source term as used in NUREG–1465, which was written primarily to address loss of coolant accidents (LOCA). The regulatory guidance for this rule extends the NUREG–1465 source terms to other accidents which involve core damage. The definition suggested by NEI is consistent with the proposed use of the AST. The § 50.2 definition has been revised in the final rule to reflect the change suggested by NEI and that suggested by Arizona Public Service Comment 1 below.

5. NEI Comment 4

NEI stated that the proposed rule does not permit new test reactors to use an alternative source term. New test reactors would have to use the Part 100 Subpart A, ''Evaluation Factors for Stationary Power Reactor Site Applications Before January 10, 1997, and for Testing Reactors,'' even though their application for an operating license would be filed after January 10, 1997. The use of Section 50.67, ''Accident Source Term,'' is limited to holders of operating licenses issued before January 10, 1997. This wording prohibits new test reactors from using the alternative source term. NEI recommended that § 50.67 be amended to allow new test reactors to use an alternative source term.

NRC Response. Section 50.67 applies only to holders of licenses for operating reactors, including test reactors, whose licenses were issued before January 10, 1997. There is no regulatory requirement for a specific source term for reactors to be licensed in the future, including test reactors. Accordingly, no regulatory action is necessary to accommodate the NEI recommendation.

6. Duke Energy Corporation Comment

Duke Energy Corporation (Duke) endorsed the comments submitted on behalf of the industry by NEI. Duke stated that the proposed § 50.67(b)(1) was not clear regarding whether licensees will be allowed to use a revised source term on a limited basis (e.g., for analyses of a specific accident or function), or whether they will be required to review the entire radiological consequence analyses to apply for the new source term. Duke suggested that necessary guidance be provided in the draft regulatory guidance to allow for limited use of the new source terms where such use can be justified.

NRC Response. This comment is similar to NEI Comment 1 addressed previously. As stated in the SOC, the NRC will consider justifiable limited (i.e., selective) applications of an AST. Although a selective implementation may result in a reduced scope of evaluation, the licensee must still demonstrate that the AST implementation and any associated proposed modifications will not exceed the criteria specified in § 50.67. The scope of the required re-analyses will depend on the specific application proposed by the licensee. Regulatory guidance on selective implements and the scope of required re-analyses has been included in the draft guide and are available as announced in this **Federal Register** notice.

7. Arizona Public Service Company Comment 1

Arizona Public Service Company (APS) noted that the SOC statement, ''a subsequent change to the source term must be made through a license amendment'' could be interpreted as requiring prior NRC approval for any change in the magnitude and mix of radionuclides released from the reactor core. APS stated that this interpretation could place additional restrictions on licensee efforts at economical fuel management, including reload design.

NRC Response. The NRC agrees with the APS comment. The NRC had intended the phrase ''magnitude and mix'' to refer to the fractions of the fission product inventory of the radionuclides released from the reactor fuel. The NRC intent for the provision in question was to require approval for changes in the radioactivity release fractions, the radionuclides released, their physical and chemical form, and the timing of their release. Since ''magnitude and mix'' could be a source of confusion, the NRC has modified the § 50.2 definition of Source Term in the final rule to read: ''Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.'' This is consistent with NUREG–1465 when it refers to ''magnitude and mix,'' since the NUREG–1465 presents these data in the form of tables of release fractions and radionuclides. This revised language also addresses NEI Comment 3 above.

8. Arizona Public Service Company Comment 2

In its second comment, APS noted that NUREG–1465 contains a disclaimer that the accident source terms provided therein may not be applicable to fuel irradiated in excess of 40 GWD/MTU. The NRC has licensed core designs with fuel irradiations of up to 62 GWD/MTU. APS questioned whether the NRC staff was going to address the affect of high burnups on a generic basis, or on a facility-by-facility basis.

NRC Response. The AST tabulated in the draft regulatory guidance, which differs in some aspects from that provided in NUREG–1465, is applicable to peak rod average irradiations up to 62 GWD/MTU. Attachment 1 to the regulatory analysis for this rulemaking describes the bases of this extension in fuel irradiation as it applies to the AST. There are some facility-by-facility considerations. For example, the increase in core inventory for some long-lived radionuclides and the change in isotopic mix due to the increase in plutonium fission as the fuel ages is addressed by the Draft Guide-1081 provision that licensees re-analyze the core inventory based on current operating parameters, including fuel burnup.

III. Section-by-Section Analysis

A. Section 50.2

The general ''definitions'' section for Part 50 is 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. The definition of source term in § 50.2 embodies the NUREG–1465 definition; however, the § 50.2 definition includes the clarifying phrase, ''expressed as fractions of the fission product inventory in the fuel,'' (see prior response to Arizona Public Service Comment 1). 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 guidance that supports this final rule describes an acceptable basis for defining the characteristics of an alternative source term.

B. Section 50.67(a)

This paragraph defines the licensees that may seek to revise their current radiological source term with an alternative source term. The final rule is applicable to holders of operating licenses that were issued under 10 CFR Part 50 before January 10, 1997, and to holders of renewed licenses issued under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997. The final rule does 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 final rule does not explicitly define an alternative source term. In lieu of an explicit reference to NUREG–1465, Footnote 1 to the final rule identifies the significant attributes of an accident source term. The regulatory guidance that is being issued to support this final rule will identify ASTs (based on the NUREG–1465 source terms) that are acceptable alternatives to the source term in TID–14844, and will provide implementation guidance. This approach will provide for future revised source terms if they are developed and will allow licensees to propose additional alternatives for NRC consideration.

C. Section 50.67(b)(1)

This paragraph of § 50.67 identifies 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 final rule have indicated that there are potential changes to the facility as documented in the FSAR that will constitute a no significant hazards consideration. However, these determinations will have to be made for each proposed change based upon facility-specific evaluations. The procedural requirements for processing a license amendment are presented 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 [3] that allows a licensee to make changes to the facility as described in the final safety analysis report (FSAR) without prior NRC approval, if the proposed change meets certain criteria specified in § 50.59. If the criteria are not met, the licensee must request NRC approval of the change using the license amendment process detailed in § 50.90. Significant to this final rule is the criterion that NRC review is required if the proposed change would result in a greater than minimal 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 might 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 fractions of the fission product inventory of the radionuclides released from the reactor fuel, their chemical and physical form, or the timing of their release as tabulated in the regulatory guidance (with deviations proposed by the licensee and approved by the NRC) could not be implemented under § 50.59. This provision applies only to these tabulated parameters.

The final rule will 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, ''Clarification of TMI Action Plan Requirements,'' requirements II.B.2, II.B.3, III.D.3.4. The regulatory guidance that supports this final 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

[3] Section 10 CFR 50.59 is being amended in a parallel, but separate, rulemaking action. That rulemaking, when implemented is expected to replace the unreviewed safety question (USQ) concept. Further, the criteria for consequences are being revised from ''may be increased'' to ''result in more than a minimal increase.'' Those changes are not expected to invalidate the conclusions drawn in this analysis.

performed again as a prerequisite for approval of the use of an alternative source term. Nor is it the NRC's intent that EAB, LPZ, and control room dose calculations be performed for all applications under § 50.67. 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 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 replacing 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 is included in § 50.67 as well as in 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 final 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 has not included 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 final rule. However, the NRC does believe that these provisions offer a useful perspective that supports the conclusion that the organ doses implied by the 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 final rule supersede the dose criteria in GDC–19, the other provisions of GDC–19 remain applicable.

There may be technically justifiable implementations of an AST that would not require calculation of the EAB, LPZ, or control room doses. For example, a proposed modification to change the closure time of a containment isolation valve from 2 seconds to 5 seconds may be based on the timing insights of the AST. Although a specific calculation might not be necessary in this case, the licensee is still required to affirm with reasonable assurance that the doses would comply with these stated criteria.

E. 10 CFR Part 50, Appendix A, GDC–19

GDC–19 is changed to include the TEDE dose criterion for control room design for applicants for construction permits, design certifications, and combined 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 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 AP600 advanced LWR in order to use the 0.05 Sv (5 rem) TEDE criterion deemed necessary for use with alternative source terms. Exemptions will arguably be necessary for future applicants for construction permits, design certifications, and combined licenses. This amendment will 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 are 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 has been added to define what constitutes an accident source term. This new footnote is identical to the existing footnote 1 to § 100.11, and was added to provide for consistency between Parts 50 and 100.

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

These paragraphs are 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 three classes of applicants. The first affected class is comprised of applicants for design certification under Part 52, Subpart B. Section 52.47(a)(1)(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 were modified to delete the reference to TID–14844. This change makes it clear that applicants for combined licenses should not use the TID–14844 source term but should 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 comprised of applicants for combined licenses under Part 52, Subpart C. Section 52.79(b) makes the requirements of 52.47(a)(1)(i) applicable if a certified design is not referenced. Thus, the combined license applicant is also subject to the requirements of Section 50.34(f).

The third 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.

IV. Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission is issuing for public comment a draft of a guide planned for its Regulatory Guide Series. This series has been developed to describe and make available to the public information such as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. Copies of the draft guide may be obtained as described in Section VI, ''Referenced Documents,'' of these statements of consideration. You may also download copies from the NRC's interactive rulemaking forum website through the NRC home page (http://ruleforum.llnl.gov/cgi-bin/rulemake).

The draft guide, temporarily identified by its task number DG–1081 (which should be mentioned in all correspondence concerning this draft guide) is titled ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' This guide is intended for Division 1, ''Power Reactors.'' This draft guide is being developed to provide regulatory guidance on the implementation of an alternative source term at an operating reactor. The guide addresses 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 provides guidance on the scope and extent of affected design basis accident (DBA) radiological analyses and associated acceptance criteria. The guide includes revised assumptions and methods for each affected DBA in a series of appendices. These appendices supersede the guidance in Regulatory Guides 1.3, 1.4, 1.5, 1.25, and 1.77, and supplement guidance in Regulatory Guide 1.89 for those facilities using an alternative source term.

The draft guide has not received complete NRC staff review and does not represent an official NRC staff position.

Previous draft versions of DG–1081 have been made publicly available to support technical interactions with the public. This **Federal Register** announcement provides an opportunity for the public to provide comments on the DG–1081 guidance. The NRC staff will consider the public comments in its efforts to finalize the regulatory guidance.

The Commission invites advice and recommendations on the content of the draft regulatory guide. Comments and suggestion are particularly requested on the following questions.

A. Scope of Implementation

1. The guidance provided in the draft regulatory guide is intended to allow licensees the maximum flexibility in pursuing technically justifiable AST implementations provided that a clear, consistent, and logical design basis is maintained. Comments are specifically requested on the following questions.

A. Does the proposed guidance provide the desired flexibility while providing reasonable assurance that a clear, consistent, and logical design basis will be maintained?

B. Is there a less complex alternative approach that would provide the desired flexibility while maintaining a clear, consistent, and logical design basis?

C. Should the Commission allow licensees that have received approval for a selective implementation to extend the AST and the TEDE criteria to other design basis applications (that do not involve reanalysis of the DBA LOCA) under § 50.59 rather than under § 50.67 as currently proposed?

2. The guidance would allow selective implementation of the characteristics (i.e., the fractions of fission product inventory of the radionuclides released from the reactor fuel, their chemical and physical form, and the timing of their release) of an AST. The Commission believes that implementations based only on the timing insights of an AST may be technically justifiable. The Commission believes that the other combinations may be internally inconsistent. Comments are specifically requested on the following questions.

A. What other combinations of AST characteristics are technically consistent?

B. What plant modifications might be based on these combinations?

B. Scope of Re-Analyses

1. The draft regulatory guide provides guidance on the scope of the re-analyses that should be performed to support an AST implementation. Comments are requested on the following questions.

A. Is the proposed guidance on the scope of re-analyses technically appropriate and clear? How could it be improved?

B. The guidance allows licensees to disposition certain impacts of an AST on the basis of the NRC staff's re-baselining study. Does this study or other documents provide a sufficient basis for the Commission to generically disposition these impacts?

2. It may be possible for licensees to demonstrate that the doses from certain affected analyses assessed using the prior source term and dose methodology would be greater than the doses obtained using a proposed AST and the TEDE methodology. The proposed guidance would allow the licensee to disposition these affected analyses without re-calculation. Nonetheless, the design basis would now include the approved AST and TEDE criteria. The guidance in the draft regulatory guide would require the licensee to update the calculation to be consistent with the approved AST and dose methodology described in the facility design basis in

the event of a subsequent re-calculation. Comments are requested on the following questions.

A. Should the Commission allow licensees to continue to use the prior source term and dose criteria for these analyses and not require that they be updated on subsequent revisions?

B. If the analyses are not updated, how will licensees assure that the earlier conclusion that the analyses are limiting remains valid following subsequent revisions?

3. Analyses of the integrated radiation doses for environmental qualification of certain equipment important to safety will be affected by the increased concentration of radioactive cesium in the containment sump water. The Commission has been considering the position that licensees proposing to implement an AST must address all impacts of the proposed implementation, including the impact of the increased cesium concentration. However, the Commission now believes it may be necessary for all operating power reactors to address the postulated increase in the cesium concentration. The Commission will consider this issue as a generic safety issue. Comments are requested on the following questions.

A. Is there information that should be considered by the Commission in resolving this generic issue?

B. If the Commission should conclude that there is safety significance but that the costs of implementing corrective actions are not justified on a generic basis, should licensees who are voluntarily proposing to amend their design basis to use an AST be required to address the impact of the increased cesium concentration?

C. If a licensee proposes a change in the plant configuration that would result in an increase in the integrated dose for one or more components and this licensee is also proposing, or has already implemented an AST, should the re-analysis of the integrated dose be based on that AST or on the prior TID14844 source term?

Comments may be accompanied by relevant information or supporting data. Written comments may be mailed to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff. Mail Stop O16C1. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by March 7, 2000.

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://ruleforum.llnl.gov/cgi-bin/rulemake). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415–5905; or by internet electronic mail to cag@nrc.gov. For information about the draft guide, contact Mr. Stephen F. LaVie, (301) 415–1081; Internet electronic mail sfl@nrc.gov.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

V. Draft Standard Review Plan Section; Issuance, Availability

The Nuclear Regulatory Commission is issuing for public comment a draft of a new section to NUREG–0800, ''Standard Review Plan.'' Standard review plan (SRP) sections are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. The draft SRP Section 15.0.1, is titled ''Radiological Consequence Analyses Using Alternative Source Terms.'' The SRP section complements draft regulatory guide DG–1081. The draft SRP section has not received complete NRC staff review and does not represent an official NRC staff position.

Copies of the draft SRP section may be obtained as described in Section VI, ''Referenced Documents,'' of these statements of consideration. You may also download copies from the NRC's interactive rulemaking forum website through the NRC home page (http://ruleforum.llnl.gov/cgi-bin/rulemake).

Comments on the content of the draft SRP section are invited. Comments may be accompanied by relevant information or supporting data. Comments should be submitted as described above for the draft regulatory guide. Although a time limit is given for comments on this draft SRP section, comments and suggestions in connection with items for inclusion in SRP sections currently being developed or improvements in all published SRP sections are encouraged at any time.

VI. Referenced Documents

Copies of NUREG–0737, NUREG–0800, NUREG–1465, NUREG/BR–0058, NUREG/BR–184, 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.

Single copies of regulatory guides, both active and draft may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington DC 20555–0001, or by fax to (301) 415–2289, or by email to distribution@nrc.gov. Active guides may also be purchased from the National Technical Information Service on a standing order basis. Details of this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161. Copies of active and draft guides are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington DC.

Copies of SECY–94–302, SECY–96–242, SECY–98–154, SECY–98–289, TID–14844, 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. 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 final rule allows 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 are not 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. Those plant changes that do not require prior NRC review and approval pursuant to § 50.59 are not likely to involve any significant increase in environmental impacts. The § 50.59 criteria are sufficiently stringent that any potential change in plant design that could have an adverse environmental impact in all likelihood could not be made by the licensee without prior NRC review and approval. Every plant change that requires NRC review and approval under § 50.59 requires a license amendment and, therefore, the preparation of an environmental assessment to determine whether the proposed change involves any significant environmental impact. Thus, this final rule, by itself, will not result in plant changes that involve any significant increase in environmental impacts. The final rule does not affect non-radiological plant effluents.

The NRC requested public comments on any environmental justice considerations that may be related to this rule. No public comments relevant to the draft environmental assessment or environmental justice considerations were received. The NRC requested the views of the States on the environmental assessment for this rule. No comments relevant to the draft environmental assessment or environmental justice considerations were received.

The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 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 final 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 relative to the total burden estimated, 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 Act 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 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 final rule, and that a backfit analysis is not required for this rulemaking because these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1). This final rule amends the NRC's regulations by establishing alternate requirements that may be voluntarily adopted by licensees, and makes changes to the regulations to conform them to a 1996 rulemaking.

XII. Small Business Regulatory Enforcement Fairness Act

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

XIII. National Technology Transfer and Advancement Act

The National Technology Transfer Act of 1995, Pub. L. 104–113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule the NRC is establishing a government-unique standard in Section 50.67(b)(2) by specifying accident radiation dose criteria. These criteria were issued for use by future license applicants by an earlier rulemaking (61 FR 65157, December 11, 1996) and, by this final rule, are being applied to operating reactors that voluntarily use an alternative source term. No voluntary consensus standard has been identified that could be used instead of the government-unique standard.

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 and 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 the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as 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 [11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, 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 [11] 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 [11] 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.

* * * * *

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, and holders of renewed licenses under part 54 of this chapter whose initial operating license was 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

[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.

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).

(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, section II, ''Protection by Multiple Fission Product Barriers,'' ''Criterion 19—Control room'' is revised to read as follows:

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

* * * * *

II. Protection by Multiple Fission Product Barriers

* * * * *

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 and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, 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 17th day of December 1999.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 99–33283 Filed 12–22–99; 8:45 am]

BILLING CODE 7590–01–P

[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] 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.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

RIN 3150–AG23

AP600 Design Certification

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC or Commission) is amending its regulations to certify the AP600 standard plant design under Subpart B of 10 CFR part 52. This action is necessary so that applicants or licensees intending to construct and operate an AP600 design may do so by referencing this regulation [AP600 design certification rule (DCR)]. The applicant for certification of the AP600 design was Westinghouse Electric Company LLC (hereinafter referred to as Westinghouse).

EFFECTIVE DATE: The effective date of this rule is January 24, 2000. The incorporation by reference of certain documents listed in this regulation is approved by the Director of the Office of the Federal Register as of January 24, 2000.

FOR FURTHER INFORMATION CONTACT: Jerry N. Wilson, Mail Stop O–12 G15, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or telephone (301) 415–3145, or e-mail: *jnw@nrc.gov*.

SUPPLEMENTARY INFORMATION:

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