RULEMAKING ISSUE
(Affirmation)

October 5, 1999 SECY-99-240

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: FINAL AMENDMENTS TO 10 CFR PARTS 21, 50, AND 54 AND AVAILABILITY FOR PUBLIC COMMENT OF DRAFT REGULATORY GUIDE DG-1081 AND DRAFT STANDARD REVIEW PLAN SECTION 15.0.1 REGARDING USE OF ALTERNATIVE SOURCE TERMS AT OPERATING REACTORS

PURPOSE:

To obtain the Commission's approval to publish a final rule to amend 10 CFR Parts 21, 50, and 54 to provide for the use of alternative source terms (ASTs) at operating reactors and obtain the Commission's approval to announce the availability for public comment of draft regulatory guide DG-1081, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and draft Standard Review Plan (SRP) Section, 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

SUMMARY:

This paper and the accompanying attachments present for the Commission's approval a final rule to amend 10 CFR Parts 21, 50, and 54. These amendments set forth requirements and acceptance criteria for the use of a revised source term as an alternative to the TID-14844 source term by operating reactors. Operating reactors would have the option of continuing to use the TID-14844 source term or an approved alternative, such as that given in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." This action would allow interested licensees to pursue cost-beneficial licensing actions to reduce unnecessary regulatory burden without compromising the margin of safety of the facility. Many of the AST

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applications may provide concomitant improvements in overall safety and in reduced occupational exposure.

The amendment also makes conforming changes to eliminate the need for exemptions from certain requirements in 10 CFR 50.34(f) and from the dose criterion in 10 CFR Part 50, Appendix A, General Design Criterion (GDC-19), by applicants for construction permits, combined operating licenses, or design certifications after January 10, 1997.

In support of this final rule, the staff prepared a new draft regulatory guide DG-1081 and a new draft Section 15.0.1 of the Standard Review Plan (NUREG-0800). An announcement of the availability of the draft guide and SRP section for public comment is included in the *Federal Register* notice (FRN) for the final rule (Attachment 1).

BACKGROUND:

In SECY-96-242, dated November 25, 1996, the staff described its proposed approach for allowing licensees of operating reactors to voluntarily amend their facility design basis to use the revised source terms provided in NUREG-1465. The Commission approved the staff's approach in a staff requirements memorandum (SRM) dated February 12, 1997. In SECY-98-154, dated June 30, 1998, the staff reported on the results of its rebaselining analysis of the potential impacts of the revised source terms. In SECY-98-158, dated June 30, 1998, the staff provided a rulemaking plan for the proposed rulemaking. The Commission directed the staff to proceed with expedited rulemaking in an SRM dated September 4, 1998.

The final rule and the draft regulatory guidance prepared to support the rulemaking addresses the use of ASTs in lieu of endorsing the revised source terms of NUREG-1465. The draft regulatory guide prepared to support the rulemaking defines the acceptable source term assumptions for operating reactors and tabulates the significant characteristics of an acceptable AST. Specific cross-references to NUREG-1465 were not made. This protocol (1) allows licensees to propose technically justifiable alternatives to the NUREG-1465 source terms, (2) facilitates adapting the NUREG-1465 source terms to design basis accidents (DBAs) other than a loss-of-coolant accident (LOCA), and (3) allows the staff to adjust the NUREG-1465 data to address extended burnup fuels.

In SECY-98-289, dated December 15, 1998, the staff requested the Commission's approval to publish proposed revisions to 10 CFR Parts 21, 50, and 54 to provide for the use of ASTs at operating reactors. The proposed rulemaking would also make conforming changes to eliminate the need for exemptions from certain requirements in 10 CFR 50.34(f) and from the dose criterion in 10 CFR Part 50, Appendix A, GDC-19, by applicants for construction permits, combined operating licenses, or design certifications after January 10, 1997. The Commission approved publication of the proposed rule in an SRM dated February 25, 1999. The proposed rule was published in the Federal Register on March 11, 1999. The SRM also directed the staff to prepare a regulatory guide (Attachment 2) and an SRP section (Attachment 3) to provide regulatory guidance in support of the rule.

In addition to the documents previously addressed herein, this package includes the final regulatory analysis (Attachment 4), the final environmental assessment and finding of no significant impact (Attachment 5), draft congressional letters (Attachment 6), and a draft public announcement (Attachment 7).

This rulemaking is being tracked as Item VI.I in the August 25, 1998, EDO response to issues raised within the Senate authorization context and the stakeholder meeting of July 17, 1998 (Chairman's Tasking Memorandum).

DISCUSSION:

As previously noted, the proposed rule was published in the *Federal Register* on March 11, 1999, for a 75-day public comment period, which expired on May 27, 1999. Comments received through June 25, 1999, were considered in preparing the final rule. Seven comment letters were received. The commenters included two State agencies, two nuclear industry groups, and three nuclear utilities. Commenters were supportive of the proposed rule. The Nuclear Energy Institute (NEI) suggested a change in the definition of *source term* provided in 10 CFR 50.2 to better represent its applicability to DBAs other than a LOCA. The other comments suggested clarification of the Statements of Consideration or the draft regulatory guide. The disposition of the comments is discussed in the FRN for the final rule (Attachment 1).

The staff has conducted two public meetings. In the first meeting on April 20, 1999, the staff distributed copies of the working draft of the accident analysis appendices of the draft regulatory guide to facilitate technical interactions and discussed the staff's then-current thinking on the regulatory positions to be included in the draft guide. In the second meeting on June 2, 1999, the staff distributed copies of the working draft of the guide (with appropriate disclaimers regarding its unofficial status). This version was also posted on the agency's rulemaking technical conferences website. Meeting minutes, including all presentation materials, were placed in the agency's Public Document Room (PDR). During these meetings, the NEI representatives provided technical information on the content of the draft guide. All of the information provided was considered by the staff. Because of time constraints, the staff has not been able to fully evaluate some of the information provided. The staff intends to evaluate this information further during the public comment period for the draft regulatory guide.

The staff placed a copy of this final rulemaking package in the PDR on July 5, 1999, the date that the document was distributed to the Advisory Committee on Reactor Safeguards (ACRS).

Two changes were made to the proposed rule language in preparing this final rule. In the first change, the definition of the term *source term* proposed for 10 CFR 50.2 was revised from the following:

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

to read:

> *Source term* refers to the fractions of the fission product inventory of the radionuclides released from the reactor fuel, their physical and chemical form, and the timing of their release.

Although the language in the proposed rule is consistent with the definition in NUREG-1465, the final definition is closer in intent to the use of the AST as provided in the draft regulatory guide.

In the second change, the applicability of the final 10 CFR 50.67, was revised from the following:

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

to read:

> *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.

This change was made to correct the unintentional exclusion of holders of renewal licenses.

In the process of developing this final rule, the draft regulatory guide, and the draft SRP section, the staff has determined that an aspect of the AST should be addressed as a generic safety issue. In typical environmental qualification integrated dose calculations, it has been traditionally assumed that 50 percent of the core inventory of radioiodine, no noble gases, and 1 percent of the inventory of remaining radionuclides would be in the containment sump water. The source terms in NUREG-1465 identify that 30 to 40 percent of the core inventory of radioiodines and 25 to 30 percent of the core inventory of cesium could be available for transfer to the sump water. Since the radioactive decay half-life of cesium-137 is about 30 years as compared to 8.3 days for iodine-131, the increased presence of cesium will increase the integrated dose to components exposed to radiation from the sump water over the longer periods associated with environmental qualification. On the basis of the insights of the ASTs with regard to cesium releases, the staff believes that it is necessary to consider the potential impact of the larger cesium concentration in the containment sump water as it applies to all licensees of operating power reactors, not just to those licensees amending their design basis to use an AST. The staff will pursue resolution of this issue as a generic safety issue to determine whether regulatory action is justified. The staff included several questions related to this issue in the proposed notice of availability for the draft guide and solicited specific comments on it. The staff expects to resolve this issue in parallel with the finalization of the regulatory guide, and does not expect that it will be necessary to revise the final rule. In the interim period before final resolution of this issue, the staff will consider the TID-14844 source term to be acceptable in reanalyses of the impact of proposed plant modifications on previously analyzed integrated component doses regardless of the accident source term used to evaluate offsite and control room doses.

The staff has considered the potential impact of the postulated cesium concentration on the operability of safety systems at currently operating reactors. Staff analyses have shown that the EQ doses determined using the current TID-14844 source term are more limiting than those calculated using the NUREG-1465 source terms for exposure periods less than about 30 days to four months following the accident. The postulated increase in the cesium concentration is not a concern for those systems and components having a safety function that is performed and completed earlier than thirty days following an accident. The staff concludes that continued plant operation does not pose an immediate threat to public health and safety since this equipment will remain capable of performing its intended design functions.

Equipment having a safety function that is performed for periods longer than 30 days, such as long-term cooling, would initially operate as designed but may be adversely affected if the postulated integrated radiation dose exceeded design criteria. The GSI will evaluate the validity of these concerns and determine whether further regulatory action is required. The staff determined that should such equipment fail there will not be an undue threat to public health and safety. The staff based its decision on (1) the onsite and offsite emergency response organizations would have been activated and protective measures for members of the public implemented for the events that could result in EQ doses of the magnitude being considered, and (2) the availability of time between the onset of the event and the projected failure provides sufficient time to take compensatory measures to prevent the equipment failure or to restore the degraded safety function.

A regulatory analysis (Attachment 4) was prepared to evaluate the value and impact of the final rulemaking. This analysis concludes that the public health and safety and the common defense and security would continue to be adequately protected. The analysis qualitatively determined that the potential values associated with the revised source term are substantial enough to justify the rulemaking.

An environmental assessment (Attachment 5) was performed, and it was determined that the issuance of the final rule is not a major Federal action that significantly affects the quality of the human environment. Therefore, an environmental impact statement is not required. The actual accident sequence and progression are not changed; it is the regulatory assumptions regarding the accident that will be affected by the change. The use of an AST alone cannot increase the core damage frequency (CDF), the large early release frequency (LERF), actual offsite or onsite radiation doses, or other non-radiological impacts. However, an AST could be used to justify changes in the plant design that might have an impact on the CDF or the LERF or might increase offsite or onsite doses. These potential changes are subject to existing requirements in the Commission's regulations, such as 10 CFR 50.59 and 10 CFR 50.91, and the associated potential environmental impacts would not be significantly increased. Thus, the protection of public health and safety is not decreased.

The staff has determined that the backfit rule, 10 CFR 50.109, does not apply to this final regulation, and, therefore, a backfit analysis is not required. 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 Commission's regulations by establishing alternate requirements that may be voluntarily adopted by licensees.

RESOURCES

In response to a request in the SRM to SECY-98-158 of September 4, 1998, the NRC staff prepared an estimate of the resource implications of this final rule. The analysis is documented in the regulatory analysis. In summary, the NRC staff projects that efforts to review license amendment requests submitted under the final rule will require 0.8 full-time equivalents (FTE) in FY 2000 and 1.2 FTEs in both FY 2001 and FY 2002. Resources for the review of license amendments are currently budgeted.

COORDINATION:

OGC has no legal objection to the content of this paper. OCFO has reviewed this Commission paper for resource implications and has no objection to its content. The OCIO has reviewed this final rule for information technology and information management implications and concurs in it. The Office of Management and Budget has approved an NRC determination that the information collection burden associated with the final rule was insignificant. The NRC staff did not coordinate this rulemaking with the Agreement States because this rule is only applicable to licensees regulated by the NRC in accordance with Part 50. The ACRS and CRGR were briefed on this final rule on August 10, 1999 and August 25, 1999, respectively, and have no objection to its publication. The ACRS and CRGR have reviewed the draft regulatory guide and the draft SRP section and have no objection to their publication for public comment.

RECOMMENDATION:

That the Commission --

1. Approve the notice of final rulemaking for publication (Attachment 1).

2. Certify that this rule, if promulgated, will have no negative economic impact on a substantial number of small entities to satisfy the requirements of the Regulatory Flexibility Act, 5 U.S.C. 605(b).

3. Note that:

   a. The final rule (Attachment 1) would be published in the *Federal Register* and made effective 30 days later;

   b. The Notice of Availability of the draft regulatory guide and the draft SRP section for a 75-day public comment period is included in the FRN for the final rule;

   c. The public comment period for the draft regulatory guide and the draft SRP section was extended from the 45 days as provided for in the rulemaking plan (SECY-98-158, dated June 30, 1998) to 75 days;

   d. The staff will provide the final regulatory guide and SRP section to the EDO by May 1, 2000;

e. The final regulatory analysis will be available in the PDR (Attachment 4);

f. The final environmental assessment and the finding of no significant impact have been prepared (Attachment 5);

g. As required by the Regulatory Flexibility Act, a regulatory flexibility analysis has been prepared and is part of the *Federal Register* notice. The evaluation indicates the economic impact on licensees and small entities will not be significant. The Chief Counsel for Advocacy of the Small Business Administration will be notified of the Commission's determination;

h. This final rule was determined to involve an increase in information collection requirements. Because the burden for this information collection is insignificant relative to the total burden estimated for other license amendment requests under § 50.90, Office of Management and Budget (OMB) clearance is not required.

i. The appropriate congressional committees will be informed (Attachment 6);

j. A public announcement will be issued (Attachment 7); and

k. Copies of the FRN of Final Rulemaking will be distributed to all Commission power reactor licensees. The notice will be sent to other interested parties upon request.

L. The final rule uses a government-unique standard instead of a voluntary consensus standard as no voluntary consensus standard has been identified that could have been used.

William Travers

William D. Travers
Executive Director
for Operations

Attachments:
1. Federal Register Notice of Final Rule
2. Draft Regulatory Guide (DG-1081)
3. Draft Standard Review Plan Section 15.0.1
4. Final Regulatory Analysis
5. Final Environment Assessment and Finding of No Significant Impact
6. Draft Congressional Letters / Congressional Review Act Forms
7. Draft Public Announcement

Commissioners' completed vote sheets/comments should be provided directly to the Office of the Secretary by COB Wednesday, October 20, 1999.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT October 13, 1999, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of November 1, 1999. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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Attachment 1

Federal Register

Notice of Final Rulemaking

[7590-01-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: *<30 days from publication date>*

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

An applicant for an operating license is required by § 50.34(b) to submit a final safety analysis report (FSAR) that describes the facility and its design bases and limits, and presents a safety analysis of the structures, systems, and components of the facility as a whole.

Guidance in performing these analyses is given in regulatory guides. In its review of the more recent applications for operating licenses, the NRC has used the review procedures in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP). These review procedures reference or provide acceptable assumptions and analysis methods. The facility FSAR documents the assumptions and methods actually used by the applicant in the required safety analyses. The NRC's finding that a license may be issued is based on the review of the FSAR, as documented in the Commission's safety evaluation report (SER). 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

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[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.

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 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

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[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

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.

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or different from a previously applicable staff position.

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 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 is 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.

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 issue 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, by substituting the phrase, "fraction of the fission product inventory of the radionuclides released from the reactor fuel." 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.

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 uses the term, "fractions of the fission product inventory of the radionuclides released," as a substitute for NUREG-1465's "magnitude and mix," (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 proposed 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 proposed 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.

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[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.

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 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 proposed rule. However, the

NRC does believe that these provisions offer a useful perspective that supports the conclusion that the organ doses implied by the 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 be 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 operating licenses that submitted applications after January 10, 1997 (the effective date of the 1996 rulemaking adopting the TEDE criterion), and for those licenses using an alternative source term under § 50.67. The 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 operating 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 operating 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://www.nrc.gov).

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 (insert date 75 days from publication of this notice.)

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking 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://www.nrc.gov).

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-0184, 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 fractions of the fission product inventory of the radionuclides released from the reactor fuel, 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)

*   *   *   *   *

[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.

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 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.

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

been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

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

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

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

* * * * *

II. * * *

Criterion 19 — Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

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

* * * * *

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

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

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

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

§ 54.4 **Scope.**

(a) * * *

(1) * * *

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

* * * * *

Dated at Rockville, Maryland, this ____ day of ________ 1999.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,
Secretary of the Commission.

Attachment 2

Draft

Regulatory Guide

(DG-1081)

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

September 1999
Division 1
Draft DG-1081

DRAFT REGULATORY GUIDE

Contact: S.F. LaVie (301)415-1081

DRAFT REGULATORY GUIDE DG-1081

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. 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

Requests for single copies of draft or active regulatory guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section, or by fax to (301)415-2289; or by email to DISTRIBUTION@NRC.GOV.

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APPENDICES

A. INTRODUCTION

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information," requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, "Reactor Site Criteria," Section 100.11,[1] "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Ref. 1), is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref. 2). The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility final safety analysis report (FSAR). Fundamental assumptions that are design inputs, including the source term, are required to be included in the FSAR and become part of the facility design basis.[2]

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. A holder of an operating license issued prior to January 10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise the accident source term used in design basis radiological consequence analyses.

This guide is being developed to provide guidance on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. The effective regulatory guide will establish an acceptable alternative source term (AST) and

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[1] Applicants for a construction permit, a design certification, or a combined license who applied after January 10, 1997, are required to meet radiological criteria provided in 10 CFR 50.34.

[2] As defined in 10 CFR 50.2, *design bases* means information that 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 or experiments or both) 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 a range of values) for controlling parameters that constitute reference bounds for design.

identify the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide would also identify acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

In general, information provided by regulatory guides is reflected in NUREG-0800, the Standard Review Plan (SRP) (Ref 3). The NRC staff uses the SRP to review applications to construct and operate nuclear power plants. This regulatory guide will apply to Chapter 15.0.1 of the SRP.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

B. DISCUSSION

An accident source term is intended to be 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 staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. The design basis accidents (DBAs) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to address known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired 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 (Ref. 4) 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 (Ref. 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff 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. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The

NRC staff also determined that some licensees might wish to use an AST in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to usze an AST[3] in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and this regulatory guide.

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67. Specifically, the superseded regulatory guides are:

Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" (Ref. 6)

Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (Ref. 7)

Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors" (Ref. 8)

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Ref. 9)

Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Ref. 10)

The guidance in Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants." regarding the radiological source term used in the determination of integrated doses for environmental qualification purposes is superseded by the corresponding guidance in this regulatory guide for those facilities that are proposing to, or have already implemented an AST. All other guidance in Regulatory Guide 1.89 remains effective.

This draft guide primarily addresses design basis accidents, such as those addressed in Chapter 15 of typical final safety analysis reports (FSAR). This guide does not address all areas of potentially significant risk. Although this guide addresses fuel handling accidents,

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[3]The NUREG-1465 source terms have often been referred to as the "revised source terms." In recognition that there may be additional source terms identified in the future, 10 CFR 50.67 addresses "alternative source terms." This regulatory guide endorses a source term derived from NUREG-1465 and provides guidance on the acceptable attributes of other alternative source terms.

other potential events that could occur during shutdown operations are not currently addressed. The NRC staff has several ongoing initiatives involving risks of shutdown operations, extended burnup fuels, and the risk-informing current regulations. The information in this guide may be revised in the future as NRC staff evaluations are completed and regulatory decisions on these issues are made.

C. REGULATORY POSITION

1. IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

1.1 Generic Considerations

As used in this guide, an AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and that has been approved for use under 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and identifies significant characteristics of other ASTs that may be found acceptable. While the NRC staff recognizes several potential uses of an AST, it is not possible to foresee all possible uses. The NRC staff will allow licensees to pursue technically justifiable uses of the ASTs in the most flexible manner compatible with maintaining a clear, logical, and consistent design basis. The NRC staff will approve these license amendment requests if the facility, as modified, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs.

1.1.1 Safety Margins

The proposed uses of an AST and the associated proposed facility modifications should be evaluated to determine whether the proposed changes are consistent with the principle that a sufficient safety margins are maintained, including margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval.

1.1.2 Defense in Depth

The proposed uses of an AST and the associated proposed facility modifications should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.

Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.

1.1.3 Integrity of Facility Design Basis

The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff will authorize technically justifiable partial, or selective, uses of an AST if a clear, logical design basis is maintained. Sensitivity analyses may be able to show that existing analyses are adequately conservative and re-calculation is not warranted. This approach would create two tiers of analyses, those based on the previous source term (and found to be bounding) and those based on an AST. The radiological acceptance criteria would be different with some analyses based upon whole body and thyroid criteria and some based on TEDE criteria. The facility design bases should clearly indicate that the source term assumptions and radiological criteria in these earlier analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.

Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses, radiological and non-radiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.

1.1.4 Emergency Preparedness Applications

Requirements for emergency preparedness at nuclear power plants are set forth in 10 CFR 50.47, "Emergency Plans." Additional requirements are set forth in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. The planning basis for many of these requirements was published in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."[4] This joint effort by the Environmental Protection Agency (EPA) and the NRC considered the principal characteristics (such as nuclides released and distances) likely to be involved for a spectrum of design basis and severe (core melt) accidents. No single accident scenario is the basis of the required preparedness. The objective of the planning is to provide public protection that would bound a wide spectrum of possible events with a sufficient basis for extension of response efforts for unanticipated events. These requirements were issued after a long period of involvement by numerous stakeholders, including the Federal Emergency Management Agency, other Federal agencies, local and State governments (and in some cases, foreign governments), private citizens, utilities, and industry groups.

[4] This planning basis is also addressed in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Ref. 13).

Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient *by itself* as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

This guidance does not, however, preclude the appropriate use of the insights of the AST in establishing emergency response procedures such as those associated with emergency dose projections, protective measures, and severe accident management guides.

1.2 Scope of Implementation

The AST described in this guide is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. The accident source term is a fundamental assumption upon which a large portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. A complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of the AST. However, the NRC staff has determined that there could be implementations for which this level of re-analysis may not be necessary. Two categories are defined: Full and selective implementations.

1.2.1 Full Implementation

A full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. A full implementation replaces the previous accident source term used in all design basis radiological analyses and establishes the TEDE dose criteria. At a minimum, the DBA LOCA must be re-analyzed using the guidance in Appendix A of this guide. Additional guidance on analysis is provided in Regulatory Position 1.3 of this guide. After a full implementation is approved, all subsequent new or updated analyses should be based on the AST and TEDE criteria. Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, Tests and Experiments," or unless the new application involved a change to a technical specification. However, a change from an approved AST to a different AST which is not approved for use at that facility, would require a license amendment under 10 CFR 50.67.

1.2.2 Selective Implementation

A selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees the maximum flexibility in technically justified selective implementations provided a clear, logical, and consistent design basis is maintained. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. In this case, the licensee may only need re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory Position 1.3 of this guide. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. If the licensee desires to use the AST and TEDE criteria in a different application, another license amendment under 10 CFR 50.67 would be required.

1.3 Scope of Required Analyses

1.3.1 Design Basis Radiological Analyses

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following.

- Equipment Environmental Qualification (10 CFR 50.49)
- Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50)
- Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)
- Alternative Source Term (10 CFR 50.67)
- Environmental Reports (10 CFR Part 51)
- Facility Siting (10 CFR 100.11)[5]

There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to the following from Reference 2, NUREG-0737.

- Post-accident Access Shielding (NUREG-0737, II.B.2)
- Post-accident Sampling Capability (NUREG-0737, II.B.3)
  Accident Monitoring Instrumentation (NUREG-0737, II.F.1)
- Leakage Control (NUREG-0737, III.D.1.1)
- Emergency Response Facilities (NUREG-0737, III.A.1.2)
- Control Room Habitability (NUREG-0737,III.D.3.4)

1.3.2 Re-analysis Guidance

Any implementation, full or selective, of an AST and any associated facility modification is expected to be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-evaluation will necessarily be a function of the specific

[5] Dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for licensees that have implemented an AST.

proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

The NRC staff has performed an evaluation of the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 1) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, evaluations may need to be performed regarding the ability of the damper to close against increase containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

For a full implementation, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated in accordance with the guidance in this section.

A selective implementation of an AST and any associated facility modification based on the AST is expected to evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST and, if dose calculations are performed, the TEDE criteria. The licensee may not implement other characteristics of the AST or extend the AST to other plant modifications without prior NRC

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[6] For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of DBA LOCA doses, re-assessment of the containment pressure and temperature transient, recalculation of sump pH, re-assessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.

approval under 10 CFR 50.67. For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be necessary if the modified elapsed time remains a small fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual case basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.

1.3.3 Use of Sensitivity or Scoping Analyses

It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a *sensitivity analysis* is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied. A *scoping analysis* is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose.

1.3.4 Updating Analyses Following Implementation

A full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria will be addressed in all affected analyses on an individual as-needed basis. Re-evaluation using the previously approved source term may not appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility do not constitute a change in analysis methodology that would require NRC approval.

This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. Use of the approved AST and TEDE criteria in analyses that were not in the scope of the approved implementation requires prior NRC staff approval under 10 CFR 50.67.

1.3.5 Equipment Environmental Qualification

Current EQ analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts.

1.4 Risk Implications

The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact on the existing PRAs should be evaluated.

Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.

The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. For guidance, refer to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15).

1.5 Submittal Requirements

According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref 16), provides additional guidance. The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses, since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and non-radiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.

If the licensee has used a current approved version of a NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

1.6 FSAR Requirements

Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.

2. ATTRIBUTES OF AN ACCEPTABLE ALTERNATIVE SOURCE TERM

An acceptable accident source term is not set forth in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an accident source term that is acceptable to the NRC staff for use at operating power reactors. A substantial effort was expended by the NRC, its contractors, various national laboratories, peer reviewers, and others, in performing severe accident research and in developing the source terms provided in NUREG-1465 (Ref.5). However, future research may identify opportunities for changes in these source terms. The NRC staff will consider applications for an AST different from that identified in this guide. However, the NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. To be considered acceptable, an alternative source term must have the following attributes:

2.1 The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

2.2 The AST must be expressed in terms of times and rates of appearance of radioactive fission products the into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine.

2.3 The AST must not be based upon a single accident scenario but should instead represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

2.4 The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and documented in a scrutable form that facilitates public review and discourse.

2.5 The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.

3. ACCIDENT SOURCE TERM

This section provides an alternative accident source term acceptable to the NRC staff. The data in Regulatory Positions 3.2 through 3.5 are fundamental to the definition of an AST. Once approved, the data in these positions become part of the facility's design basis. Deviations from this guidance must be evaluated against Regulatory Position 2. Once the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.

3.1 Core Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and 1.02 times the current licensed rated thermal power.[7] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[8] For non-LOCA events, the appropriate radial peaking factor from the facility's core operating limits report (COLR) should be applied. No adjustment to the core inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions[9]

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage states for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for

[7] The uncertainty factor used in determining the core inventory should be that value provided in Paragraph I.A. of Appendix K to 10 CFR Part 50.

[8] Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel burnup. Thus, the maximum inventory at the end of life should be used.

[9] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The NRC has ongoing research activities regarding the impact of extended burnup on fuel releases. These release fractions may be revised in the future pending the outcome of these activities. These data may not be applicable to cores containing mixed oxide (MOX) fuel.

PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 1
BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase
Noble Gases	0.05	0.95
Halogens	0.05	0.25
Alkali Metals	0.05	0.20
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

Table 2
PWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase
Noble Gases	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3.[10] These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

[10] The differences in magnitude between the gap releases shown in Tables 1 and 2 and the gap fractions shown in Table 3 compensate for uncertainties in the extrapolation of release data evaluated for LOCA events to the non-LOCA events. The NRC previously assumed 0.1 for all noble gases and halogens, and 30% for Kr-85. The fractions shown in Table 3 are consistent with available data for extended burnup fuel (based on the limiting assembly).

Table 3
Non-LOCA
Fraction of Core Inventory in Gap

Group	Fraction
I-131	0.12
Kr-85	0.15
Other Noble Gases	0.10
Other Halogens	0.10
Alkali Metals	0.10

3.3 Timing of Release Phases

Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[11] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and/or the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

Table 4
LOCA Release Phases

	PWRs		BWRs	
Phase	**Onset**	**Duration**	**Onset**	**Duration**
Gap Release	10-30 sec	0.5 hr	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr	0.5 hr	1.5 hr

For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes, or on the basis of an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.

3.4 Radionuclide Composition

Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.

[11] In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.

Table 5
Radionuclide Groups

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se
	Ba, Sr, Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr
	Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

3.5 Chemical Form

Of the radioiodine released from the fuel, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent as elemental iodine, and 0.15 percent as organic iodide. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melting and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nuclear boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

4. DOSE CALCULATIONAL METHODOLOGY

The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results.

4.1 Offsite Dose Consequences

The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):

4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should include all radionuclides that are significant with regard to dose consequences and the released radioactivity.[12]

4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 17). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 18), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield the CEDE.

4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.47×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.32×10^{-4} cubic meters per second.

4.1.4 The DDE should be calculated using submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining TEDE. Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 19), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield the EDE.

4.1.5 The TEDE should be determined for the most limiting receptor at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is reported. The timing of the increments should be consistent with the rate at which analysis parameters change.

4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.

4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.

4.2 Control Room Dose Consequences

The following guidance should be used in determining the TEDE for persons located in the control room:

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[13] The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.

4.2.1 The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:

- Contamination of the control room atmosphere by the intake or infiltration of the radioactivity contained in the radioactive plume released from the facility,

- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactivity from areas and structures adjacent to the control room envelope,

- Radiation shine from the external radioactive plume released from the facility,

- Radiation shine from the reactor containment,

- Radiation shine from radioactivity in systems and components inside or external to the control room envelope, e.g., radioactivity buildup in recirculation filters.

4.2.2 The radioactivity releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values, unless these assumptions would result in non-conservative results for the control room.

4.2.3 The models used to transport radioactivity into and through the control room,[13] and the shielding models used to determine radiation shine from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Credit for engineered safety features that mitigate airborne activity within the control room may be assumed. Such features may include control room isolation or pressurization, intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 23), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs. Several aspects of RMs can delay the isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

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[14] The iodine protection factor (IPF) methodology of Ref. 20 may not be adequately conservative for all DBAs and control room arrangements and should only be used with caution. The NRC computer codes HABIT (Ref. 21) and RADTRAD (Ref. 22) incorporate suitable methodologies.

4.2.5 Credit should generally not be taken for personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[14] For the duration of the event, the breathing rate of this individual should be assumed to be 3.47×10^{-4} cubic meters per second.

4.2.7 Control room doses should be calculated using dose conversion factors identified above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is represented by a hemisphere of volume, V, in cubic feet.

$$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173} \qquad \text{Equation 1}$$

4.3 Other Dose Consequences

The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, all radiation exposures to plant personnel should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.

4.4 Acceptance Criteria

In 10 CFR Part 50, § 50.67 establishes the radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

The acceptance criteria for the various NUREG-0737 items generally reference GDC-19 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67.b.2.iii

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14 This occupancy is modeled in the χ/Q values determined in Ref. 20 and should not be credited twice. The ARCON96 Code (Ref. 24) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

Table 6[15]
Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria
LOCA	25 rem TEDE
BWR Main Steam Line Break	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
BWR Rod Drop Accident	6.25 rem TEDE
PWR Steam Generator Tube Rupture	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
PWR Main Steam Line Break	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
PWR Locked Rotor Accident	2.5 rem TEDE
PWR Rod Ejection Accident	6.25 rem TEDE
Fuel Handling Accident	6.25 rem TEDE

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency

[15] For PWRs with steam generator alternate repair criteria, different dose criteria may apply to SGTR and MSLB analyses.

power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of maximizing the postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but may be nonconservative in many cases when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications[16]. If a range of values or a tolerance band is specified, the value that would maximize the postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

5.1.4 Applicability of Prior Licensing Basis

The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review to staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, Backfitting. However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 Accident-Specific Assumptions

The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The

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[16] Note that for some parameters, the technical specification value may be superseded for analysis purposes by values provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 23) and in Generic Letter 99-02 (Ref. 25) rather than the surveillance test criteria in the technical specifications.

DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency.

The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.

5.3 Meteorology Assumptions

Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 20, and 26).

References 20 and 26 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable and should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 27) implements Regulatory Guide 1.145 (Ref. 26) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[17] (Ref. 24) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be

[17] The ARCON96 computer code contains processing options that may yield χ/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release point and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.

used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 28). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.

6. ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION

The assumptions in Appendix I to this guide should be used for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.

D. IMPLEMENTATION

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

This draft guide has been released to encourage public participation in its development. Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods to be described in the active guide incorporating public comments will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.

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APPENDIX A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT

The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs). These assumptions supplement the guidance provided in Draft Regulatory Guide DG-1081.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

Source Term Assumptions

1. Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.

2. If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

Assumptions on Transport in Primary Containment

3. Assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows:

3.1 The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to assure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.

3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the SRP, NUREG-0800 (Ref. 1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. 2). The latter model is incorporated into the analysis code RADTRAD (Ref. 3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

3.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. 1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP[1] and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[2] (Ref. 4). This simplified model is incorporated into the analysis code RADTRAD (Refs. 1-3).

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

3.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. 5 and 6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.

3.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool. Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.

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[1] The SRP establishes a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. Since the activity is now assumed to be released continuously over a period of time, the maximum activity needs to be redefined. If the release from the fuel is to be modeled as a linear ramp over the duration of the release phase, the maximum activity should be the activity remaining at the end of the early in-vessel phase. If the release from the fuel is assumed to occur as a step increase at the start of the early in-vessel release phase, maximum activity should be the activity assumed to be released at that time.

[2] This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those which will maximize the dose consequences. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects the iodine species assumed by the model to be present initially.

3.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Chapter 6.5.4 of the SRP (Ref. 1).

3.7 The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced, but not less than 50% of the maximum leak rate, after the first 24 hours if the reduced leak rate is supported by plant configuration and analyses. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.

3.8 For BWRs with Mark III containments, the flow rate from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This flow rate should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.

3.9 If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.

Assumptions on Dual Containments

4. For facilities with dual containment systems, the assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.

4.1 Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the release point is more than two and one-half times the height of any adjacent structure.

4.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.

4.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95%, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%), of the total numbers of hours in the data set.

4.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.

4.5 Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas filled lines may be considered on a case-by-case basis.

4.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. 5) and Generic Letter 99-02 (Ref. 6).

Assumptions on ESF System Leakage

5. Engineered safety feature (ESF) systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. 8). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions should be used in evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.

5.1 With the exception of noble gases, all the fission products released from the fuel should be assumed to be instantaneously and homogeneously mixed in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models that transport containment airborne activity to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.

5.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. 9), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., ECCS pump miniflow return to the refueling water storage tank.

5.3 In addition to the leakage specified in paragraph 5.2, the evaluation should assume leakage from a gross failure of a passive component at the rate of 50 gallons per minute, starting 24 hours after the accident and lasting for 30 minutes. This evaluation is not required if the facility design provides an ESF ventilation filtration system that exhausts the areas of potential leakage.

5.4 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.

5.5 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

Where: h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.

5.6 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.

5.7 The radioiodine that is postulated to become airborne should be assumed 97% elemental and 3% organic and should be assumed to be released to the environment. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. 5) and Generic Letter 99-02 (Ref. 6).

Assumptions on Main Steam Isolation Valve Leakage in BWRs

6. For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions should be used for evaluating the consequences of MSIV leakage.

6.1 For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage. No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

6.2 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced, but not less than 50% of the maximum leak rate, after the first 24 hours if the reduced leak rate is supported by site-specific analyses.

6.3 Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes rather than on slug flow, but other models may be used if justified.

6.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release. Holdup and dilution in the turbine building should not be assumed.

6.5 Reduction in MSIV releases that are due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. Reference 10 provides guidance on acceptable models.

Assumption on Containment Purging

7. The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If applicable, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.

APPENDIX A REFERENCES

A-1 USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." NUREG-0800

A-2 Powers, D.A., et al, "A simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," NUREG/CR-6189, 1995

A-3 USNRC, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, 1998

A.4 Powers, D.A., and Burson, S.B., "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, 1993

A.5 USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Regulatory Guide 1.52

A.6 USNRC, "Laboratory Testing of Nuclear Grade Activated Charcoal," Generic Letter 99-02, June 3, 1999

A-7 Powers, D.A., "A Simplified Model of Decontamination by BWR Steam Suppression Pools," NUREG/CR-6153, 1996

A-8 USNRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991

A-9 USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, 1980

A-10 Cline, J.E., "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991

Appendix B

ASSUMPTIONS USED FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

This appendix provides assumptions acceptable to the staff for evaluating the radiological consequences of a fuel handling accident at light water reactors. These assumptions supplement the guidance provided in the main body of this guide.

1. Source Term

Assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply.

1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered. Chapter 15.7.4, "Radiological Consequences of Fuel Handling Accidents," of the SRP (Ref. B-1) contains an example of a conservative bounding analysis.

1.2 The gap activity fractions of Table 3 in Regulatory Position 3 of Draft Regulatory Guide DG-1081 should be assumed. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.

1.3 Of the radioiodine released from the fuel, 99.75% of the released iodine should be assumed to be in the form of elemental iodine and 0.25% in organic species.

2. Water Depth

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental and organic iodine species results in the iodine above the water being composed of 45% elemental and 55% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-2).

3. Noble Gases

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

4. Fuel Handling Accidents Within the Fuel Building

For fuel handling accidents postulated to occur within the fuel building, the following assumptions should be made.

4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.

4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-3, B-4). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system[1] should be determined and accounted for in the radioactivity release analyses.

4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building.

5. Fuel Handling Accidents Within Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions should be made:

5.1 If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.

5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.

5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

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[1] These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

[2] Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in technical specifications.

[3] Administrative controls are to be established to close the airlock or hatch in less than 30 minutes. These controls generally state that a dedicated individual must be present at the open airlock or hatch while fuel handling operations are in progress and that this individual must have any necessary equipment to close the airlock or hatch in the required time. Radiological analyses should generally not credit this manual isolation.

5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-3 and B-4). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]

5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume.

APPENDIX B REFERENCES

B-1. USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." NUREG-0800

B-2. Burley,G., "Evaluation of Fission Product Release and Transport," 1971

B-3. USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Regulatory Guide 1.52

B-4. USNRC, "Laboratory Testing of Nuclear Grade Activated Charcoal," Generic Letter 99-02, June 3, 1999

Appendix C

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR ROD DROP ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod drop accident at BWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of Draft Regulatory Guide DG-1081. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 1 and the percentage of the fuel affected.

2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by the technical specifications. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the NSSS vendor's standard technical specifications.

2.1 The concentration that is the maximum value (typically 4 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike and

2.2 The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.

3. The assumptions related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows.

3.1 The activity released from the fuel from either the gap or from fuel melt is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.

3.3 Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.

3.4 Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground-

[1] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

level release at a rate of 1% per day[2] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

3.5 In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses accounts for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.

3.6 The release from the reactor coolant within the pressure vessel should be assumed to consist of 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.

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[2] If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.

Appendix D

ASSUMPTIONS USED FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line accident at BWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.

Source Term

1. Assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this Draft Regulatory Guide DG-1081. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 1 and the percentage of the fuel affected.

2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the NSSS vendor's standard technical specifications.

2.1 The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike; and

2.1 The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.

3. The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.

Transport

4. The assumptions related to the transport, reduction, and release of radioactive material to the environment are as follows.

4.1 The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.

4.2 The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.

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[1] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

4.3 All the radioactivity in the released coolant should be assumed to be released to the atmosphere within 2 hours as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.

4.4 The iodine release from the main steam line should be assumed to consist of 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.

Appendix E

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

Source Terms

1. Assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 of DG-1081and the percentage of the fuel affected. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.

2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.

2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).

2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.

3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," for acceptable assumptions and methodologies for performing radiological analyses.

[2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

4. The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases via the steam generators should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

Transport[3]

5. The assumptions related to the transport, reduction, and release of radioactive material to the environment are as follows.

5.1 For facilities that have not implemented alternative repair criteria (ARC) (See Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

5.5 All iodine and particulate radionuclides released from the primary system via the faulted steam generators should be assumed to be released to the environment with no mitigation.

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[3]In this appendix, *Ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. *Faulted* refers to the state of the steam generator in which the secondary side has been depressurized due to a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred. *Partitioning Coefficient* is defined as:

$$P.C. = \frac{\text{mass of } I_2 \text{ per unit mass of liquid}}{\text{mass of } I_2 \text{ per unit mass of gas}}$$

5.6 During periods of total submergence of the tubes in the non-faulted steam generators, the primary-to-secondary leakage should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

5.7 A partitioning coefficient for elemental iodine of 100 should be assumed during periods of total submergence of the tubes in the nonfaulted steam generators. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

5.8 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partititioning. The impact of emergency operating procedure restoration strategies on steam generator water level need to be considered.

APPENDIX E REFERENCES

E-1 USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

E-2 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988

Appendix F

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

Source Term

1. Assumptions regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this draft regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.

2.1 A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).

2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.

3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (Ref. F-1), for acceptable assumptions and methodologies for performing radiological analyses.

[2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

4. Iodine releases via the steam generators should be assumed to be 97% elemental and 3% organic.

Transport[3]

5. The assumptions related to the transport, reduction, and release of radioactive material to the environment are as follows:

5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

5.6 During periods of total submergence of the tubes in the ruptured steam generator, the transport model described in this section should be utilized for iodine and particulates. This model is shown in Figure F-1 and summarized below:

- A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant.

- The leakage that flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR

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[3] In this appendix, *Ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. *Partitioning Coefficient* is defined as:

$$P.C. = \frac{\textit{mass of } I_2 \textit{ per unit mass of liquid}}{\textit{mass of } I_2 \textit{ per unit mass of gas}}$$

Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. F-2).

- The leakage that does not flash is assumed to mix with the bulk water and will become vapor at a rate that is the function of the steaming rate and the partition coefficient.

- A partitioning coefficient for elemental iodine of 100 may be assumed during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

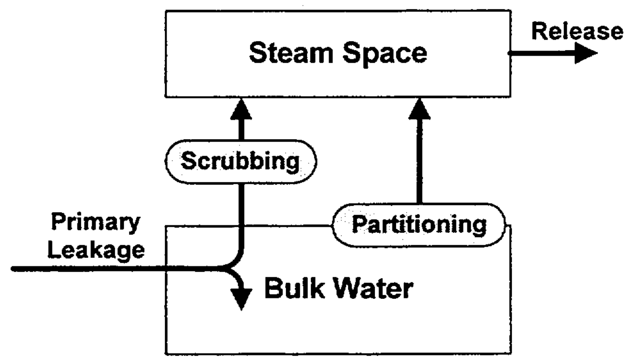

Figure F-1
Transport Model

5.7 For the non-ruptured steam generators used to perform post-event cooldown, primary coolant leakage should be assumed to mix with the bulk water without flashing. A partitioning coefficient for elemental iodine of 100 may be assumed during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators, as discussed above.

5.8 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. F-3). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level need to be considered.

APPENDIX F REFERENCES

F-1. USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

F-2. USNRC, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409

F-3. USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988

Appendix G

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a locked rotor accident at PWR light water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide.

Source Term

1. Assumptions regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of this guide and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break and steam generator tube rupture.

3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

4. The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases via the steam generators should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

Release Transport

5. The assumptions related to the transport, reduction, and release of radioactive material to the environment are as follows.

5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on

[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (Ref. G-1), for acceptable assumptions and methodologies for performing radiological analyses.

lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity should continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

5.6 During periods of total submergence of the steam generator tubes, the primary-to-secondary leakage should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

5.7 A partitioning coefficient[2] of 100 should be assumed for elemental iodine during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

5.8 Operating experience and analyses have shown that, for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. G-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level needs to be considered.

APPENDIX G REFERENCES

G-1. USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

G-2 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988

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[2] Partitioning Coefficient is defined as:

$$P.C. = \frac{\text{mass of } I_2 \text{ per unit mass of liquid}}{\text{mass of } I_2 \text{ per unit mass of gas}}$$

Appendix H

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod ejection accident at PWR light water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide.

Source Term

1. Assumptions regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.

3. Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85& elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the LOCA event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

5. Iodine releases via the steam generators should be assumed to be 97% elemental and 3% organic.

Transport From Containment

6. The assumptions related to the transport, reduction, and release of radioactive material in and from the containment are as follows.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (Ref. H-1), for acceptable assumptions and methodologies for performing radiological analyses.

6.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating LOCAs.

6.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.

Transport From Secondary System

7. The assumptions related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows.

7.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.

7.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

7.3 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

7.4 During periods of total submergence of the steam generator tubes, the primary-to-secondary leakage should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

7.5 A partitioning coefficient[2] of 100 should be assumed for elemental iodine during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

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[2] Partitioning Coefficient is defined as:

$$P.C. = \frac{\text{mass of } I_2 \text{ per unit mass of liquid}}{\text{mass of } I_2 \text{ per unit mass of gas}}$$

7.6 Operating experience and analyses have shown that, for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. H-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level needs to be considered

APPENDIX H REFERENCES

H-1 USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

H-2 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988

Appendix I

ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION

This appendix addresses assumptions associated with equipment qualification that are acceptable to the NRC staff for performing radiological assessments. As stated in Regulatory Position 6 of DG-1081, this appendix supersedes Regulatory Positions 2.c.(1) and 2.c.(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. I-1), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in this appendix, other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.

Basic Assumptions

1. Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) activity suspended in the containment atmosphere, (2) activity plated out on containment surfaces, and (3) activity mixed in the containment sump water. A given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

Fission Product Concentrations

2. The radiation environment resulting from normal operations should be based on the conservative source term estimates reported in the facility's Safety Analysis Report or should be consistent with the primary coolant specific activity limits contained in the facility's technical specifications. The use of equilibrium primary coolant concentrations based on 1% fuel cladding failures would be one acceptable method.

3. The radioactivity released from the core during a design basis loss-of-coolant accident (LOCA) should be based on the assumptions provided in Regulatory Position 3 and Appendix A of this regulatory guide. Although the design basis LOCA is generally limiting for radiological environmental qualification (EQ) purposes, there may be components for which another design basis accident may be limiting. In these cases, the assumptions provided in Appendices B through G of this regulatory guide, as applicable, should be used. The other appendices to this regulatory guide identify facility features and natural phenomena that may be considered in design basis analyses. Applicable features and mechanisms may be assumed in EQ calculations provided that any prerequisites and limitations identified regarding their use are met. There are additional considerations:

- For PWR ice condenser containments, the source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the ice beds as functions of time.

- For BWR Mark III designs, all the activity should be assumed initially released to the drywell area and the transfer of activity from these regions via containment leakage to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

Dose Model for Containment Atmosphere

4. The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere and from the plateout of aerosols on containment surfaces generally should be calculated for the midpoint in the containment, and this dose rate should be used for all exposed components. Radiation shielding afforded by internal structures may be neglected since their inclusion would involve a higher degree of complexity than is warranted. It is expected that the shielding afforded by these structures would reduce the dose rates by factors of two or more depending on the specific location and geometry. More detailed calculations may be warranted for selected components if acceptable dose rates cannot be achieved using the simpler assumptions.

5. Because of the short range of the betas in air, the airborne beta dose rates should be calculated using an infinite medium model. Other models, such as finite cloud and semi-infinite cloud, may be applicable to selected components with sufficient justification. The applicability of the semi-infinite model would depend on the location of the component, available shielding, and receptor geometry. For example, beta dose rates for equipment located on the containment walls or on large internal structures might be adequately assessed using the semi-infinite model. Use of a finite cloud model will be considered on a case-by-case method.

6. All gamma dose rates should be multiplied by a correction factor of 1.3 to account for the omission of the contribution from the decay chains of the isotopes.

Dose Model For Containment Sump Water Sources

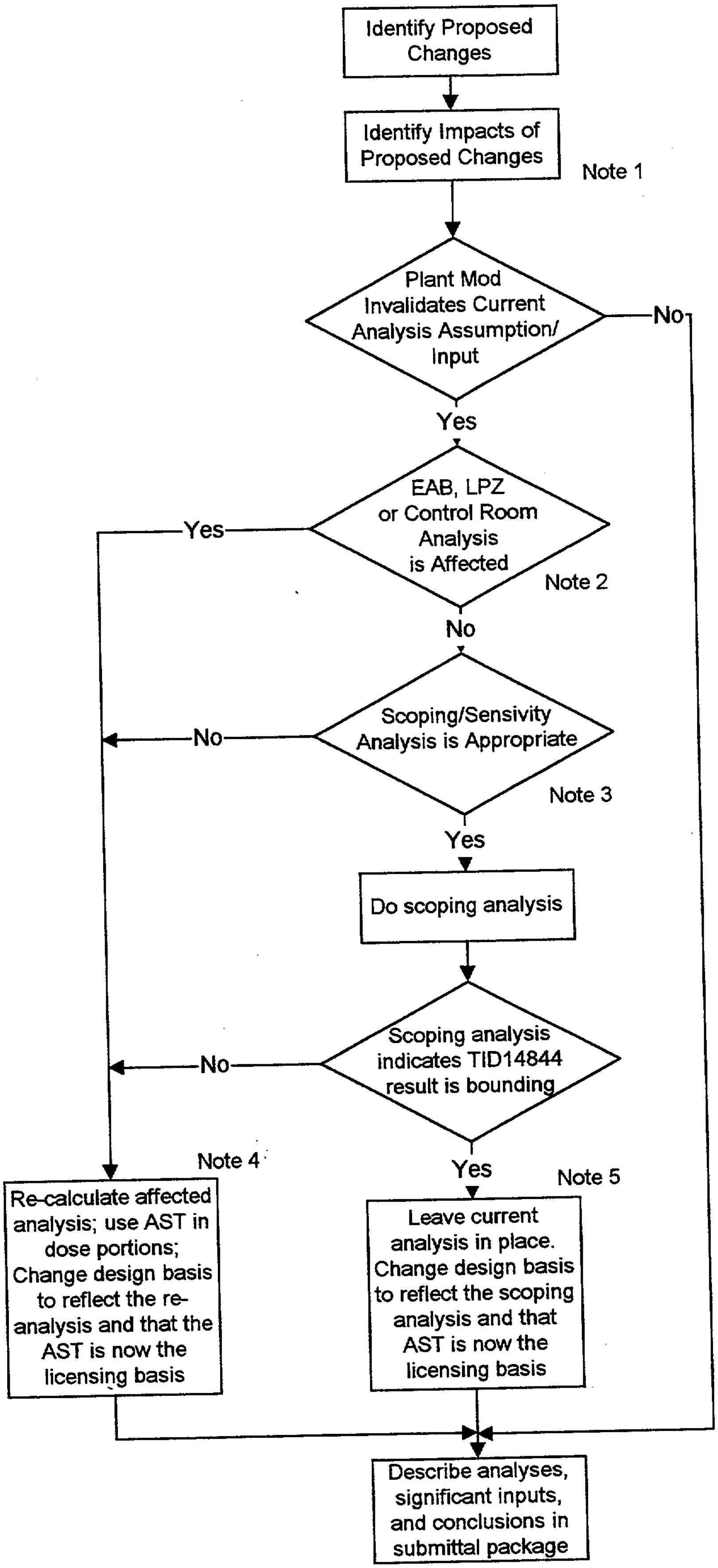
7. With the exception of noble gases, all the activity released from the fuel should be assumed to be transported to the containment sump as it is released. This activity should be assumed to mix instantaneously and uniformly with other liquids that drain to the sump. This transport can also be modeled mechanistically as the time-dependent washout of airborne aerosols by the action of containment sprays. Radionuclides that do not become airborne on release from the reactor coolant system, e.g., they are entrained in non-flashed reactor coolant, should be assumed to be instantaneously transported to the sump and be uniformly distributed in the sump water.

8. The gamma and beta dose rates and the integrated doses should be calculated for a point located on the surface of the water at the centerline of the large pool of sump water. The effects of buildup should be considered.

APPENDIX I REFERENCES

I-1. USNRC, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Regulatory Guide 1.89, 1984

Appendix J
Re-analysis Decision Chart



Note 1: All impacts, radiological and non-radiological, need to be evaluated. A full implementation will include, as a minimum, a full DBA LOCA analysis.

Note 2: Sensitivity /scoping analyses should not comprise a significant part of the exclusion area boundary (EAB), low population zone (LPZ), and control room analyses.

Note 3: Scoping analyses may be used where a number of similar analyses are involved and generic conclusions can be drawn. However, scoping analyses should not be used for EAB/ LPZ/CR doses.

Note 4: If any dose analysis is to be re-calculated, the upgrade should address the selected (or all) characteristics of the source term and, as applicable, TEDE.

Note 5: Once the design basis source term is changed from the current design basis source term to a new AST, the selected AST becomes the design basis source term for all future radiological analyses, including revisions to those analyses that were shown to be bounding with the previous source term. There is no requirement to update these later analyses unless future plant modifications invalidate one or more assumptions, making such re-analysis necessary.

VALUE / IMPACT STATEMENT

A separate draft value/impact analysis has not been prepared for this draft guide. A value/impact analysis was included in the regulatory analysis for the proposed amendments to 10 CFR Parts 21, 50, and 54 published on March 11, 1999 (64 FR 12117). This regulatory analysis was updated as part of the final amendments to 10 CFR Parts 21, 50, and 54, expected to be published in October 1999. Copies of the regulatory analyses are available for inspection or copying for a fee in the Commission's Public Document Room at 2120 L Street NW, Washington, DC, under RGIN AG12.

Appendix K

Acronyms

AST	Alternative Source Term
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CEDE	Committed Effective Dose Equivalent
COLR	Core Operating Limits Report
DBA	Design Basis Accident
DDE	Deep Dose Equivalent
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
EDE	Effective Dose Equivalent
EPA	Environmental Protection Agency
EQ	Environmental Qualification
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
IPF	Iodine Protection Factor
LERF	Large, Early Release Fraction
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
MOX	Mixed Oxide
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RMS	Radiation Monitoring System
NDT	Non-Destructive Testing
TEDE	Total Effective Dose Equivalent
TID	Technical Information Document
TMI	Three Mile Island

Attachment 3

Draft

Standard Review Plan Section

(15.0.1)

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

REVIEW RESPONSIBILITIES

Primary - various (see text)

Secondary - various (see text)

The NRC expects that most operating reactors will implement an alternative source term (AST) only as a means to justify desirable plant modifications. (In the text that follows the phrase, "implementation of an AST" includes any associated plant modifications.) These modifications may be to systems or procedures identified in the final safety analysis report (FSAR) or changes to the technical specifications. The review of the affected structures, systems, components, and accident analyses is covered in other sections of the Standard Review Plan (SRP). Those sections identify the branches responsible for the review of the modifications, as well as the acceptance criteria, areas of reviews, and evaluation documentation associated with those reviews. The review of the radiological consequences of the proposed modification as described in this SRP section is performed by the SPSB with the assistance of other technical review branches in the NRC's Office of Nuclear Reactor Regulation (NRR), as deemed necessary.

The nature of the licensee's request will determine which technical branch will serve as the primary review branch for the overall proposed amendment request. This primary review branch has overall responsibility for leading the technical review, drafting the staff safety evaluation report (SER) or other appropriate regulatory document, and coordinating input from other technical review organizations.

- Probabilistic Safety Analysis Branch (SPSB) holds the responsibility for reviewing the impact of the proposed plant modification on the radiological consequences of design basis accidents (DBAs). It assists the primary review branch by reviewing probabilistic risk analysis (PRA) information submitted by the licensee. It reviews issues related to severe accidents for operating reactors

- Reactor Systems Branch (SRXB) holds the responsibility for issues related to functional performance, design, operation, and accident response of the reactor core and reactor thermal-hydraulic systems (reactor coolant systems, normal and emergency core cooling).

- Plant Systems Branch (SPLB) holds the responsibility for issues related to the functional performance, design, operation, and accident response of essential auxiliary, support, and balance-of-plant systems. It reviews issues related to design features provided to ensure protection of operators from releases of toxic and radioactive gases. It reviews issues related to design and performance of containments and their associated systems, and fuel storage and fuel handling systems.

- Mechanical and Civil Engineering Branch (EMEB) holds the responsibility for issues related to static and dynamic analysis for mechanical systems and components.

- Materials & Chemical Engineering Branch (EMCB) holds the responsibility for issues related to materials engineering, inservice inspection, and materials integrity related aspects of design and performance of reactor components and systems. It reviews issues related to chemical engineering, including containment sump pH, and containment spray performance for radioiodine scavenging. It reviews the impact of toxic gases on control room habitability.

- Electrical & Instrumentation and Controls Branch (EELB) holds the responsibility for issues related to the functional performance, design, and operation of onsite power systems, reactor trip systems, engineered safeguards features actuation systems, and plant instrumentation systems. It reviews environmental qualification of electrical equipment important to safety.

- Technical Specifications Branch (RTSB) develops, maintains, and updates standard technical specifications. It provides NRR interpretation of specific technical specification requirements and provides assistance in screening incoming change requests.

- Operator Licensing, Human Performance, and Plant Support (IOLB) holds the responsibility for issues related to operator licensing, in-plant radiation protection, effluent release control, and emergency preparedness. It reviews issues related to emergency operating procedures, human factors engineering design, in-plant radiation protection, and effluent release control.

- Office of Nuclear Regulatory Research (RES) assists the primary and secondary review branch, as requested, by providing necessary technical support.

I. AREAS OF REVIEW

Section 50.67, *Accident source term,* of 10 CFR Part 50 allows a holder of an operating license issued before January 10, 1997, to voluntarily revise the accident source term used in design basis radiological consequence analyses. Paragraph 50.67(b) requires that applications under this section contain an evaluation of the consequences of applicable DBAs previously analyzed in the plant's FSAR. Potential changes in consequences could be due to the impact of the characteristics of the AST itself or from the proposed plant modifications. Draft Guide 1081, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power s* (Ref. 1), provides guidance to licensees on performing evaluations and reanalyses in support of the implementation of an AST. Although, this SRP section is written primarily for the review of the application for the initial implementation of an AST, it is expected to be of use in reviewing applications for subsequent license amendment requests from these plants.

A complete recalculation of all design basis radiological consequence analyses may not be required for an application to be acceptable. However, applications should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed plant modifications in the context of the proposed AST. The scope and extent of the reanalysis effort, and the staff review, will depend on the specifics of the application. Draft Guide 1081 provides guidance on required reviews.

An AST is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. An accident source term is a fundamental assumption upon which a large portion of the plant design is based. Ideally, the licensee would update all design basis analyses based on the previous source term to reflect all five characteristics of the proposed AST. However, evaluations performed by the staff have indicated that this level of reanalysis may not be necessary for some AST implementations. There are potential implementations of an AST for which only limited reanalyses may be necessary. Some implementations may involve only one AST characteristic. Two categories of implementations, *full* and *selective*, are defined.

- A *full implementation* is a modification of the plant design basis that addresses all characteristics of an AST, that is, the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. A full implementation replaces the previous accident source term used in all design basis radiological analyses and incorporates the total effective dose equivalent (TEDE) dose criteria. Once a full implementation is approved, all subsequent new or updated analyses would be based on the approved AST and TEDE criteria.

- A *selective implementation* is a modification of the plant design basis that (1) is based on one or more of the characteristics of an AST and/or (2) reevaluates a limited subset of the design basis radiological analyses. An example of an application of selective implementation is one in which a licensee desires to use the release timing insights of an AST to increase the required closure time for a containment isolation valve by a small amount. The licensee would only need to evaluate the impacts of the delay in valve closure. Radiological consequence analyses might not be necessary. The staff's approval for an AST (and the TEDE criterion) would be limited to the particular selective implementation proposed by the licensee. If the licensee desires to use the approved AST and TEDE criteria in a different application, another license amendment submittal under 10 CFR 50.67 would be required.

The review associated with an application for the use of an AST is largely dependent on the scope and nature of the associated plant modifications being proposed. Thus, the areas of review identified in other SRP sections may be applicable and should be considered in performing the review. This SRP section covers the review by SPSB of the radiological consequences of DBAs. The review includes the following:

1. Reviews of the AST implementation to ensure that all significant radiological and nonradiological impacts have been considered. Radiological consequences that should be considered include the following:

    a. Exclusion area boundary (EAB), low population zone (LPZ), and control room habitability (10 CFR 50.67)

    b. Emergency response center habitability (paragraph IV.E.8 of Appendix E to 10 CFR Part 50)

    c. Equipment environmental qualification (10 CFR 50.49)

    d. Environmental assessments (10 CFR Part 51)

    e. Post-accident access shielding (NUREG-0737, II.B.2)[1]

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[1] Facility-specific licensing commitments may affect applicability of NUREG-0737 (Ref. 2) items.

f. Post-accident sampling capability (NUREG-0737, II.B.3)

g. Post-accident monitoring (NUREG-0737, II.F.1)

h. Leakage control (NUREG-0737, III.D.1.1)

i. Emergency response facilities (NUREG-0737, III.A.1.2)

j. Control room habitability (NUREG-0737, III.D.3.4)

2. A review of the sequence of accident events as described by the licensee to ensure that the case that maximizes the radioactivity release has been considered.

3. A review of the core inventory determined by the licensee to ensure that it is consistent with the current licensing basis rated thermal power, enrichment, and burnup.

4. A review of the models, assumptions, and parameter inputs used by the licensee for the calculation of the radiological consequences. For plants applying for, or having received, approval for the use of a full implementation of an AST, this SRP section supersedes the radiological analyses assumptions, acceptance criteria, and methodologies identified in the SRP sections listed below. Provisions related to the nonradiological analyses aspects of these SRP sections remain applicable.

   a. Section 15.1.5, *Steam System Piping Failures Inside and Outside of Containment* (PWR)

   b. Sections 15.3.3-15.3.4, *Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break*

   c. Section 15.4.8, *Spectrum of Rod Ejection Accidents (PWR)*

   d. Section 15.4.9, *Spectrum of Rod Drop Accidents (BWR)*

   e. Section 15.6.2, *Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment*

   f. Section 15.6.3, *Radiological Consequences of Steam Generator Tube Failure (PWR)*

   g. 15.6.4, *Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)*

   h. 15.6.5, *Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant System Pressure Boundary*

   i. 15.7.4, *Radiological Consequences of Fuel Handling Accidents.*

   This SRP section and the referenced Draft Guide 1081 (Ref. 1) may contain information that contradicts that provided in other SRP sections. In these cases, the most recent applicable information should be used.

5. Independent calculations by the staff, as necessary, to conclude, with reasonable assurance, that the licensee's analyses are acceptable.

6. Comparison of the doses calculated by the licensee and the staff against the appropriate exposure criteria, as stated in Section II below.

II. ACCEPTANCE CRITERIA

An application to replace the current DBA source term with an AST is acceptable if the plant, as modified, will continue to provide sufficient margin of safety with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs. The staff should allow licensees to pursue technically justifiable uses of an AST in the most flexible manner compatible with maintaining a clear, logical, and consistent design basis.

A complete recalculation of all design basis radiological consequence analyses may not be required for an application to be acceptable. However, applications should be supported by evaluations of all significant radiological and non-radiological impacts of the proposed plant modifications in the context of the proposed AST. The scope and extent of the reanalysis effort, and the staff review, will depend on the specifics of the application. The acceptance criteria below address the implementation of an AST and the supporting radiological consequence analyses. Additional acceptance criteria may be found in other applicable SRP sections. If the application is justified, in part, on risk insights, the acceptance criteria of SRP Section 19.0 (Ref. 3) apply.

In addition to the nonradiological acceptance criteria provided in other SRP sections, an acceptable implementation of an AST is required to demonstrate compliance with the following regulations:

- Section 50.49, *Environmental qualification of electric equipment important to safety for nuclear power plants*, of 10 CFR Part 50, as it relates to qualification of safety-related equipment with regard to integrated radiation dose during normal and accident conditions.

- Section 50.67, *Accident source term,* of 10 CFR Part 50, as it relates to the implementation of an AST in current operating nuclear power plants. For plants applying for, or having received, approval for the use of an AST, the radiological criteria in § 50.67 supersede the radiological criteria of Section 100.11, *Determination of exclusion area, low population zone, and population center distance,* of 10 CFR Part 100.

- General Design Criterion (GDC) 19, *Control Room*, of Appendix A to 10 CFR Part 50, as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

- Title 10, CFR Part 51, *Environmental protection regulations for domestic licensing and related regulatory functions,* as it relates to environmental assessments of radioactive material releases during normal and accident conditions.

- Paragraph IV.E.8 of Appendix E, to 10 CFR Part 50, *Emergency Planning and Preparedness for Production and Utilization Facilities*, as it relates to maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

An acceptable implementation of an AST should demonstrate compliance with plant-specific licensing commitments made in response to the NUREG-0737 (Ref. 2). Specific provisions[2] of interest to this SRP section include the following:

- NUREG-0737 II.B.2, *Post-accident Access Shielding*, as it relates to post-accident radiation exposure incurred while performing necessary plant operations outside of the control room..

- NUREG-0737 II.B.3, *Post-accident Sampling Capability*, as it relates to post-accident radiation exposure during sampling operations.

  NUREG-0737 II.F.1, *Additional Accident-Monitoring Equipment*, as it relates to the ability of the monitors to operate during and following an accident and perform the intended function in the accident environment.

- NUREG-0737 III.D.1.1, *Leakage Control*, as it relates to post-accident radiation exposure.

- NUREG-0737 III.A.1.2, *Emergency Response Facilities,* as it relates to maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

- NUREG-0737 III.D.3.4, *Control Room Habitability,* as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

An implementation of an AST is acceptable with regard to the radiological consequences of analyzed DBA if the calculated TEDE at the EAB and the LPZ outer boundaries do not exceed the exposure criteria listed in Table 1. The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Draft Guide-1081.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes specific aspects of this SRP section that are appropriate for the particular application. The review areas to be given attention and emphasis are based on (1) the material presented and its similarity to recently reviewed applications for other plants,

[2] The radiological criteria in these provisions reference GDC-19 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent, to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criterion should be updated for consistency with the TEDE criterion in 10 CFR 50.67.b.2.iii.

(2) the scope of the proposed AST implementation, that is., full or selective, (3) the nature and extent of associated plant modifications, and (4) whether the application is for an initial AST implementation or is based on a previously accepted implementation.

Table 1
Accident Dose Criteria[3]

Accident / Case	EAB and LPZ Dose Criteria
Loss of coolant accident	25 rem TEDE
Fuel handling accident	6.25 rem TEDE
BWR main steam line break	
Fuel damage or pre-incident spike	25 rem TEDE
Coincident iodine spike	2.5 rem TEDE
BWR rod drop accident	6.25 rem TEDE
PWR steam generator tube rupture	
Fuel damage or pre-incident spike	25 rem TEDE
Coincident iodine spike	2.5 rem TEDE
PWR main steam line break	
Fuel damage or pre-incident spike	25 rem TEDE
Coincident iodine spike	2.5 rem TEDE
PWR locked rotor accident	2.5 rem TEDE
PWR rod ejection accident	6.25 rem TEDE

1. An initial screening of the proposed application should be performed to establish the scope and extent of the needed review. An initial screening consists of the following steps:

   a. As discussed in Section I of this SRP section, a licensee can propose either a full or a selective implementation of an AST. The scope and extent of the review and the language of the staff SER will depend on this classification. The reviewer should ensure that the submittal clearly specifies the scope desired by the licensee.

   b. A preliminary review of the application for completeness and potential acceptability is performed to ensure that the application includes sufficient information to enable the reviewer to make an independent assessment regarding the acceptability of the proposal in terms of regulatory requirements and the protection of public health and safety. If the reviewer determines that the application is incomplete, or if the proposed changes cannot be accepted, the project manager should be consulted before continuing with the review.

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[3] For PWRs with steam generator alternate repair criteria, different dose criteria may apply to SGTR and MSLB analyses.

c. The reviewer should determine whether a precedent for the proposed change has been previously considered by the staff. These precedents may be identified by the licensee it its submittal or may be identified by the staff. Applicable precedents should be considered by the reviewer in structuring the review in the interest of maximizing staff efficiency and ensuring consistency of licensing actions.

d. The reviewer should identify whether the application should be considered as being risk informed. A *risk-informed licensing action* is defined as any licensing action that uses quantitative or qualitative risk assessment insights or techniques to provide a key component of the basis for the acceptability or the unacceptability of the proposed action. If the application is risk informed, a review by risk analysts in SPSB should be performed using the SRP Section 19.0.

e. The differences between the previous source term and an AST cannot, in and of themselves, affect the previously analyzed core damage frequency (CDF) and large early release frequency (LERF). However, the reviewer should ensure that any associated plant modification that may have an impact on CDF or on LERF is reviewed by risk analysts in SPSB.

f. A review of the proposed changes as they relate to the plant's licensing basis is performed. Areas of review include how the licensee satisfies certain basic regulatory requirements such as diversity, redundancy, defense-in-depth, safety margins, NUREG-0737 commitments and the General Design Criteria, as applicable. Review procedures related to structures, systems, and components, and nonradiological aspects of accidents in other SRP sections may be applicable. Previously approved implementations of an AST, if applicable, should be included in this review. If changes to technical specifications, exemptions from regulations, or other forms of relief are needed to implement the licensee's proposed change, reviewers should ensure that the appropriate requests accompany the application.

2. An application may be a *full* or *selective* implementation of an AST. The reviewer should consider the following in performing the review of the application:

a. A full implementation addresses all characteristics of an AST, replaces the previous accident source term used in all design basis radiological analyses, and incorporates the TEDE criteria of 10 CFR 50.67 and Section II of this SRP section. The reviewer should ensure that a complete analysis of the DBA LOCA has been performed, as a minimum. Other analyses may be necessary as described in Draft Guide 1081 (Ref. 1).

b. In a selective implementation, the licensee may opt to only implement one or more of the characteristics of an AST and may chose to use the AST only in analyses supporting limited plant modifications. The reviewer should ensure that the proposed selective implementation is technically justified and that a clear, logical, and consistent design basis is maintained. Since there are a large number of possible selective implementations, only generic review procedures can be provided. The reviewer will have to apply judgement. The following should be considered:

(1) A selective implementation on the basis of only the timing characteristic of an AST will normally be found acceptable without dose calculations, provided other impacts, if any, are adequately dispositioned. The acceptability of other combinations of AST characteristics is not as clear. The reviewer must ensure that the proposed combination is consistent. For example, it would be inconsistent to credit the chemical form as being cesium iodide (CsI) and ignore the increased cesium (Cs) release fraction.

(2) As previously discussed, a selective implementation need not involve dose calculations. If dose analyses are performed, the TEDE criteria in 10 CFR 50.67 and Section II of this SRP section become the design basis criteria for those analyses. The previous whole body and thyroid criteria would continue to apply to the analyses that were not affected by the implementation. This dichotomy may cause confusion if there are plant modifications associated with an AST implementation. For example--

  (a) A licensee is proposing to modify the standby gas treatment system (SGTS) as part of the proposed AST implementation. The particular modification would affect the LOCA and fuel handling accident analyses. What are the dose acceptance criteria for these two accident analyses? For the remaining unaffected accident analyses? For the control room?

  Answer: The acceptability of the design change is based on the TEDE criteria for the reanalyzed LOCA and FHA. As the remaining offsite and control room accident analyses are unaffected, the previous acceptance criteria continue to apply to those analyses.

  (b) In the previous example, what would be the result be if the modification was to control room habitability systems instead of the SGTS?

  Answer: Since the control room habitability criterion applies to all accident conditions, the licensee must demonstrate that control room doses will meet the TEDE criterion for all accidents. Once the application is approved, the design bases for these systems would incorporate the TEDE criterion.

In either case, the reviewer should ensure that the licensee's submittal and the staff's SER clearly identify the acceptance criteria for the accident, or the component, that will become part of the plant design basis.

c. Once an implementation of an AST is approved, the licensee may subsequently submit additional license amendment requests. The AST characteristics and the TEDE criteria incorporated into the design basis by previously approved AST applications are the bases for reviews of subsequent licensing actions. The reviewer should ensure that these subsequent requests are consistent with the design basis AST implementation.

3. The reviewer should ensure that the licensee has performed sufficient analyses to meet the staff's expectation that all significant potential impacts have been identified and evaluated. The reviewer should determine if the application adequately characterizes the radiological and nonradiological impacts of the proposed plant modifications in the context of the proposed AST. The reviewer should ensure that the analyses described by the licensee have the scope and depth to adequately evaluate the impacts of the change. All affected design basis analyses should be updated. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. Because of the wide scope of possible AST implementations, both full and selective, specific review guidance cannot be provided. However, the following aspects should be considered in performing these reviews:

   a. A complete recalculation of all design basis radiological analyses may not be required. However, all significant radiological and nonradiological impacts of the proposed plant modifications are to be evaluated in the context of the proposed AST.

   b. The NRC staff performed a rebaselining study (Ref. 4) of the implementation of an AST at operating reactors. This study may be referenced by a licensee to disposition the impacts of differences between source terms as they apply to radiation doses caused by fission product releases. The reviewer should ensure that all remaining radiological and nonradiological impacts of proposed plant modifications in the context of the proposed AST, including the impact on equipment environmental qualification, are evaluated. For example --

      (1) A licensee has proposed removing charcoal media from the standby gas treatment system (SGTS) on the basis of a full implementation of an AST. The licensee has reanalyzed the offsite and control room doses for all accidents that credited the SGTS filtration. Does the licensee need to reanalyze the environmental qualification (EQ) doses for components exposed to the containment airborne activity?

      Answer: In this case, the plant modification has no impact on the EQ doses. The licensee can reference the rebaselining study to disposition the airborne activity EQ doses.

      (2) As part of a larger AST implementation, a licensee proposes to eliminate in-containment fan cooler charcoal filter units. Offsite doses have been shown to be acceptable, but the in-containment source term and dose rates have increased. The new containment airborne source concentrations are greater than those previously assumed in several of the plant EQ calculations. What are the analysis requirements?

Answer: Those EQ calculations affected by the increased airborne source concentrations should be reanalyzed using the selected AST. This particular reanalysis requirement is driven not by the source term but by the plant modification. In addition, there are several potential non-radiological impacts, for example, the impact on the containment pressure-temperature transient, the impact on the fan of the reduced flow restriction, and so on, that may need to be considered.

(3) A licensee proposes to change the response time of a containment purge system isolation damper from 2.5 seconds to 5.0 seconds on the basis of timing characteristic of an AST. The licensee states that increases in offsite dose are insignificant since the containment will be isolated before to the onset of gap release. Are dose calculations necessary?

Answer: This is a selective implementation as only the timing characteristic is being proposed. The remaining characteristics of the AST are not being implemented. Thus, the previous analyses are not affected. Reanalyses would not normally be necessary. However, there may be other impacts that need to be considered, for example, can the damper close against the increased pressure that might exist at 5 seconds? Can the ductwork downstream of this damper withstand the increased pressure? If the damper could not close, dose calculations against the TEDE criteria would be warranted.

c. All affected analyses should be reevaluated and the applicable design bases updated.

d. If a particular analysis is to be recalculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria that will become part of the design basis should be addressed.

e. The licensee may use technically justifiable sensitivity or scoping evaluations to demonstrate that results from affected analyses calculated using the previous accident source term and previous dose criteria would bound the results obtained using the AST and the TEDE criteria. In this case, the affected analyses need not be updated. However, the reviewer should ensure that the licensee has made a commitment to update the design basis to indicate that the selected AST characteristics and TEDE criteria have superseded the prior source term and dose criteria. For example:

(1) As part of an AST implementation, the licensee has reanalyzed the dose for the most limiting component using the proposed AST and determined that the integrated dose would not increase by more that 20 percent. All of the licensee's existing EQ calculations include a designer's margin of a factor of two. Does the licensee need to re-calculate any additional EQ analyses? What is the design basis for the remaining analyses?

Answer: As long as the licensee has adequately identified the limiting case and the sensitivity analysis is sufficiently generic, the evaluation is appropriate and no further reanalysis is necessary. In this case, the licensee has been able to demonstrate that the existing analyses are bounding and would yield acceptable results if recalculated using the proposed AST. However, the design basis now incorporates the AST, and any future reanalysis or new analyses should be based on the AST, that is, the design basis source term.

(2) A licensee has proposed a full implementation of an AST but is not requesting any plant modifications. The licensee submitted an evaluation of the offsite and control room doses due to a DBA LOCA in support of this request. On the basis of its review of analyses performed in the staff's rebaselining study, the licensee has concluded that the existing LOCA analysis (based on the previous source term) remains bounding. In a sensitivity analysis, the licensee determined that multiplying the previous whole body result by a factor of 1.3 would yield a value that represents the TEDE dose. Is this an acceptable approach?

Answer: The implementation of an AST is a significant change to the design basis that should be viewed as a replacement rather than a adaptation of the earlier assumptions and methods. Although sensitivity and scoping analyses may have a minor role, the staff should expect that the offsite and control room dose analysis will largely be recalculated using the guidance of Draft Guide 1081 (Ref. 1). This position is taken to ensure that a clear, logical, and consistent design basis will be in place to support evaluations of future modifications, including safety evaluations under 10 CFR 50.59.

f. For a full implementation, a complete DBA LOCA analysis as described in Draft Guide 1081, should be performed as a minimum.

4. The reviewer should evaluate the AST proposed by the licensee against the guidance in Draft Guide 1081. Differences between the licensee's proposal and the guidance should be resolved with the licensee. Although the licensee is allowed to propose alternatives to the guidance, large amounts of staff resources were expended in developing the revised source term (Ref. 5) from which the draft guide source term was derived. Section 2.0 of Draft Guide 1081 provides generic guidance on what would be expected before the staff would approve an AST with deviations from the AST in Section 3.0 of the guide.

5. The analysis methods and assumptions used by the licensee in determining the core inventory should be reviewed to ensure that they are based on current licensing basis rated thermal power, enrichment, and burnup.

6. The following review should be performed for each radiological analysis described in the licensee's submittal:

a. The sequence of accident events described by the licensee should be reviewed to ensure that the analyzed case that maximizes the radioactivity release has been considered. This portion of the review should be coordinated with SRXB and SPLB as necessary.

b. The models, assumptions, and parameter inputs used by the licensee should be reviewed to ensure that the conservative design basis assumptions outlined in Draft Guide 1081 have been incorporated. Licensee-proposed alternatives to this guidance may be accepted if technically appropriate and of an appropriate level of conservatism. Significant departures from this guidance will warrant additional review.

c. Independent calculations should be performed as necessary to conclude, with reasonable assurance, that the applicant's analyses are acceptable. The staff's approval of the application is to be based on the licensee's docketed information. If differences are discovered between the licensee's methods and assumptions and those deemed acceptable to the staff, the reviewer should resolve the differences with the licensee. If necessary, the licensee should update the disputed assumptions and resubmit the affected analyses.

d. The radiation doses postulated for the EAB, the LPZ, and the control room are compared to the acceptance criteria in Section II of this SRP section.

7. The analyses of radiological doses associated with the applicable NUREG-0737 items identified in Section I are evaluated against the guidance provided in NUREG-0737 and in any license commitments related to these items. The dose criterion for these items is generally derived from the GDC-19 criteria. As GDC-19 has been updated to 5 rem TEDE, the dose criterion for NUREG-0737 items should also be 5 rem TEDE.

8. Evaluations of integrated radiation doses associated with equipment qualification are performed by EELB using the guidance of Regulatory Guide 1.89 (Ref. 6), supplemented by Appendix I to Draft Guide 1081.

9. Reviewers should determine that the proposed AST implementation and supporting analyses will be appropriately included in future updates to the licensee's FSAR. This task should be accomplished, if possible, through a review of revised FSAR pages submitted by the licensee. At a minimum, the submittal should summarize the projected changes to the FSAR. These updates should identify important assumptions that play an essential role in supporting the acceptability of the proposed implementation. Reviewers should verify that such assumptions are reflected by licensee commitments that are incorporated into the FSAR, technical specifications, or license conditions.

IV. EVALUATION FINDINGS

The reviewer prepares an SER or provides input to a larger SER prepared by the primary review branch. Findings of acceptability should have a consistent, scrutable basis that is derived from the information submitted by the licensee on the docket and the staff's evaluation of these data. The following information should be included as applicable to the particular AST implementation. These conclusions should be combined with the conclusions of other reviewers with regard to nonradiological aspects of the evaluation as applicable.

1. The AST implementation should be described in sufficient detail to reasonably document the approved design basis, as modified. This step is particularly important for selective

implementation applications. Cross-references to information submitted on the docket should be used when available in the interest of minimizing unnecessary repetition.

2. The regulatory mechanism, for example, 10 CFR 50.67, under which the change is being considered, is identified. Any regulatory exemptions, related technical specification changes, or licensee commitments are identified.

3. The licensee's supporting analyses and conclusions are described in sufficient detail to adequately document the design bases. Key analysis assumptions and inputs, analysis methods, and postulated doses should be included.

4. The staff's evaluation of the licensee's submittal, including the proposal, supporting evaluations, and conclusions drawn, should be described. Independent analyses prepared by the staff, if any, should be described. Essential analysis assumptions and inputs, analysis methods, and postulated doses should be included.

5. A conclusion similar to the following is to be included in the SER:

> The staff has reviewed the alternative source term (AST) implementation proposed by the <licensee> for the <facility>. The staff also reviewed the plant modifications associated with this proposed implementation. In performing this review, the staff relied upon information placed on the docket by <licensee>, staff experience in performing similar reviews and, where deemed necessary, on staff confirmatory calculations.
>
> The staff reviewed the assumptions, inputs, and methods used by <licensee> to assess the radiological impacts of the proposed plant modifications in the context of the proposed AST. The staff finds that <licensee> used analysis methods and assumptions consistent with the conservative guidance of Draft Guide 1081, with the exceptions discussed and accepted earlier in this SER. The staff finds the methods and assumptions used by <licensee> to be in compliance with applicable requirements. The staff compared the doses estimated by <licensee> to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds with reasonable assurance that the licensee's estimates of the total effective dose equivalent due to design basis accidents will comply with the requirements of 10 CFR 50.67 and the guidance of Draft Guide 1081. [As necessary, discuss NUREG-0737 items, equipment EQ].
>
> The staff finds reasonable assurance that the <facility>, as modified by this proposal, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The staff concludes that the proposed AST implementation and the associated plant modifications are acceptable.

6. For a full implementation of an AST, text similar to the following is to be included in the above conclusion:

> This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the <facility> design basis is superseded by the AST proposed by <licensee>. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in Draft Guide 1081. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the <facility> design basis.

7. For a selective implementation of an AST, text similar to the following is to be included in the conclusion:

> This licensing action is considered a selective implementation of the AST. With this approval, the selected characteristics of the AST and the TEDE criteria, if applicable, become the design basis for the <explain the boundaries of the approved implementation>[4]. This approval is limited to this specific implementation. The selected characteristics of the AST and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review under 10 CFR 50.67. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address the selected characteristics of the AST and the TEDE criteria as described in the <facility> design basis.

V. IMPLEMENTATION

The preceding material in this SRP section is intended to provide guidance to operating reactor licensees applying for approval of a proposed AST implementation regarding the staff's plans for performing reviews of these applications. Although primarily directed toward the review of the initial implementation, the staff will also use this SRP section in its review of license amendment requests following the initial implementation.

Except in those cases in which the licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in Draft Guide 1081, and herein, will be used by the staff in its evaluation of conformance with Commission regulations.

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[4] The description of the boundary should identify all structures, systems, and components, and accident analysis for which the design basis has been changed.

VI. REFERENCES

1. Draft Guide 1081, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.*

2. NUREG-0737, *Clarification of TMI Action Plan Requirements,* November 1980.

3. SRP 19.0, *Use of Probabilistic Risk Assessment In Plant-Specific, Risk-Informed Decisionmaking: General Guidance*, July 1998.

4. SECY-98-154, *Results of the Revised (NUREG-1465) Source Term Re-Baselining for Operating Reactors*, June 1998.

5. NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants,* February 1995.

6. Regulatory Guide 1.89, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*

Attachment 4

Final

Regulatory Analysis

REGULATORY ANALYSIS

REVISION OF 10 CFR PARTS 21, 50, AND 54; DRAFT REGULATORY GUIDE DG-1081; DRAFT STANDARD REVIEW PLAN (SRP) SECTION 15.0.1

Use of Alternative Source Terms at Operating Reactors

I. STATEMENT OF PROBLEM

This regulatory analysis addresses a final rule that amends 10 CFR Parts 21, 50, and 54. This rulemaking activity was initiated to enable holders of operating licenses issued before January 10, 1997, and holders of renewed licenses under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, to voluntarily amend their facility design basis to replace the current accident source term in design basis radiological consequence analyses with an alternative source term. Although this final rule is based on the accident source terms presented in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants*, the rule will refer to *alternative source term* (AST) to enable the use of a future alternative to NUREG -1465. (In this analysis, *revised source terms* refers to NUREG-1465.) The supporting DG-1081, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, presents an acceptable AST, based on NUREG-1465, for fuel burnups to 62 GWD/MTU (addressed in Attachment 1 to this regulatory analysis). This draft guide will be made available for public comment in conjunction with the publication of the final rule.

This final rule also amends 10 CFR Part 50 to eliminate the need for certain exemptions from Part 50 requirements for future applicants under Part 52. In addition to these future applicants, the conforming change to § 50.34(f) affects the small class of applicants that had a construction permit or manufacturing license pending on February 16, 1982. This final change allows this small class of applicants to use an alternative to the TID-14844 source term in showing compliance with § 50.34(f).

This regulatory analysis is presented in two parts, corresponding to the two considerations stated above.

A. Use of Alternative Source Terms at Operating Reactors

1. <u>Background</u>

a. Accident Source Term

A holder of an operating license (licensee) for a light-water power reactor was required by regulations issued by the US Nuclear Regulatory Commission (NRC) (or its predecessor, the U.S. Atomic Energy Commission) to submit a safety analysis report (or, for early reactors, a hazards summary report) in support of its license application assessing the radiological consequences of potential accidents and evaluated the proposed facility site. The NRC used this information in its evaluation of the suitability of the reactor design and the proposed site as required by 10 CFR Parts 50 and 100. Section 100.11 requires an applicant to assume (1) a fission product release from the core, (2) the expected containment leak rate, and (3) the site meteorological conditions to establish an exclusion area and a low population zone. A footnote to § 100.11 provides guidance that the fission product release be 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 references Technical Information Document (TID) 14844, *Calculation of Distance Factors for Power and Test Reactors*, published in 1962 by the U.S. Atomic Energy Commission, as a source of guidance and as a point of departure for addressing site-specific considerations. This fission product release, known as the TID-14844 accident source term, was used to evaluate the radiological consequences of design basis accidents (DBAs) to determine compliance with various requirements in 10 CFR Parts 50 and 100 in all of the operating reactors licensed to date. 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 TID-14844 source term was explicitly stated as a required design parameter for several Three Mile Island (TMI)-related requirements. The NRC considers the accident source term an integral part of the design basis because it was a significant input to a large portion of the plant design.

The NRC staff'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,* and Regulatory Guide 1.4, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors*, and in the Standard Review Plan (NUREG-0800), were developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in § 100.11. In that 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 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 containment, ventilation exhaust, and control room charcoal filters, are predicated on these postulated thyroid doses. This regulatory framework has provided a consistent analytical approach for evaluating the spectrum of potential consequences from DBAs.

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 effort started by the NRC and the industry after the accident at TMI. In 1995, the NRC published NUREG-1465, which utilized this research to provide more physically based estimates of accident source terms that could be applied to the design of future light-water power reactors. In NUREG-1465, the NRC provides a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These 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 based on similarity of chemical behavior. Where TID-14844 assumed an immediate release of the activity, the revised source terms have five release phases that are postulated to occur over several hours, with the onset of major core damage occurring after 30 minutes.

Where TID-14844 assumed radioiodine to be predominantly elemental, the revised source terms assume radioiodine to be predominantly cesium iodide (CsI), an aerosol that is more amenable to mitigation mechanisms. For DBAs, the NUREG-1465 source terms are comparable to the TID-14844 source term with regard to the magnitude of the noble gas and radioiodine release fractions. However, the revised source terms present a more representative description of the radionuclide composition and release timing. In SECY-94-302, *Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs*, the NRC determined that the first three phases (coolant, gap, and early in-vessel) are appropriate for design basis evaluations.

The NRC initiated several actions to provide a regulatory basis for operating reactors to voluntarily amend their facility design bases to enable the use of ASTs in design basis analyses. First, the NRC solicited information on how such source terms might be implemented. In November 1995, the Nuclear Energy Institute (NEI) submitted its generic framework (Electric Power Research Institute Technical Report TR-105909, *Generic Framework for Application of Revised Accident Source Term to Operating Plants*). This report and the NRC response were discussed in SECY-96-242 (November 1996). Second, the NRC initiated a comprehensive assessment of the overall impact of substituting the NUREG-1465 source terms for the TID-14844 source term at two typical facilities. This was done to evaluate the issues involved with applying these revised source terms at operating plants. In SECY-98-154 (June 1998), the NRC described the conclusions of this assessment. Third, the NRC accepted license amendment requests related to implementation of these revised source terms at a small number of pilot plants. The NRC has completed the review of one of these pilot projects and is currently reviewing two other pilots projects. Insights from these pilot plant reviews have been incorporated into the regulatory guidance that was developed in conjunction with this rule. Fourth, the NRC initiated an assessment on whether rulemaking would be necessary to allow operating reactors to use ASTs. The final rule described herein and the supporting regulatory guidance that were developed as part of this rulemaking are based on this assessment. The NRC is issuing the supporting draft regulatory guide for public comment on the same day it publishes this final rule.

b. Accident Dose Criteria and Control Room Dose Criteria

Part 50, Appendix A, General Design Criterion (GDC) 19, sets forth radiation dose criteria that are used to assess the suitability of the plant design with regard to maintaining control room habitability during DBAs. In § 100.11, the NRC presents radiation dose guidelines that are used to assess the suitability of the plant design with regard to offsite exposures during design basis events. The dose guidelines for the whole body and the thyroid and the immediate 2-hour exposure period were largely predicated upon 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 when Part 100 was originally published. However, the current dose guidelines, by focusing on doses to the thyroid and the whole body, assume that radioiodine will be the major contributor to doses. Although this may be appropriate with the TID-14844 source term, it may not be true for source terms 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, therefore, radioiodine may not always be the predominant radionuclide in an accident release. The revised source terms assume a larger number of radionuclides than did the TID-14844 source term as implemented in regulatory guidance. The whole body and thyroid dose guidelines ignored these contributors to dose.

In the period since these regulations were issued, there have been significant developments in the principles and scientific knowledge underlying standards for radiation dose limitation and assessment. These developments include not only updated scientific information on radionuclide uptake and metabolism, but also reflect changes in the basic philosophy of radiation protection. In 1991, the NRC revised 10 CFR Part 20, *Standards for Protection Against Radiation*, to reflect these developments. The accident dose guidelines in § 100.11 and GDC-19, were not changed when Part 20 was revised because the requisite revision to the licensing basis of each operating power reactor was not warranted. The standards in Part 20 include the dose quantity *total effective dose equivalent* (TEDE), which is defined as 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. It is expected that the thyroid could still be the limiting organ and that radioiodine could still be the limiting radionuclide, and that the current whole body and thyroid guidelines could provide adequate protection; however, this conclusion cannot be assured in all potential cases. The NRC staff recommended in SECY-96-242 that dose guidelines expressed in terms of TEDE be required if a licensee elects to use a revised source term. In a staff requirements memorandum dated February 12, 1997, the Commission directed the NRC staff to initiate rulemaking to incorporate TEDE into the regulations.

The dose guideline for the exclusion area boundary (EAB) in § 100.11 is specified with a 2-hour exposure period commencing immediately following the onset of the fission product release. This exposure period was predicated, in part, on the traditional source term assumption that the activity would be immediately available for release at the onset of the accident. The combination of these two assumptions resulted in the maximum postulated dose. The revised source terms postulate a release that occurs in phases, with the significant release starting after about 30 minutes and continuing for about 90 minutes (through the early in-vessel phase only). Because of this, an exposure period starting at the onset of the fission product release may not represent the limiting case. The NRC staff recommended in SECY-96-242 that dose guidelines expressed in terms of the worst 2-hour dose be considered if a licensee elects to use the revised source terms. In a staff requirements memorandum dated February 12, 1997, the Commission directed the NRC staff to incorporate the worst 2-hour dose in this rule.

2. Existing Regulatory Framework

a. Accident Source Term

The final rule for implementation of ASTs is applicable only to facilities that obtained an operating license, under 10 CFR Part 50, before January 10, 1997. The final rule also applies to facilities with a renewed license under 10 CFR Part 54 for which the initial operating license was issued prior to January 10, 1997. The regulations in this part are supplemented by those in other parts of Chapter 1 of Title 10, including Part 100. Part 100 contains language that qualitatively defines a required accident source term and contains a note to § 100.11 that discusses the availability of TID-14844. However, this note did not mandate the use of TID-14844. With the exception of § 50.34(f) that addresses additional TMI-related requirements, there are no explicit provisions in Title 10 requiring the use of the TID-14844 accident source term. Section 50.34(f) is only applicable to a limited number of construction permit and manufacturing license applications pending on February 16, 1982, and to applications under Part 52.

Regulatory Guides 1.3 and 1.4 specify the methods and assumptions acceptable to the NRC staff for assessing the consequences of design basis loss of coolant accidents (LOCAs) as required by § 100.11. These regulatory guides provide guidance involving accident source terms, much of which is derived from TID-14844. Other guides specify accident source terms either directly or by reference to Regulatory Guides 1.3 and 1.4. None of these guides, however, explicitly refer to TID-14844. The NRC publishes regulatory guides to describe methods acceptable to the NRC for implementing specific parts of the NRC's regulations. Because compliance with these guides is not required, applicants are permitted to propose alternatives for NRC consideration. Although NRC licensing reviews have been based on Regulatory Guides 1.3 and 1.4, the option for a licensee to propose alternatives has been and remains a possible regulatory mechanism to implement a source term other than the one in TID-14844.

An applicant for an operating license is required by § 50.34 to submit a final safety analysis report (FSAR) that describes the facility and its design bases and limits, including a safety analysis of the site and facility. 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. Although compliance with the SRP is not required, in practice many applicants adhere to the guidance in the interest of facilitating NRC review. Operating license applications docketed after May 17, 1982, are required in § 50.34(g) to contain an evaluation of the facility for conformance with the SRP. 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 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 of the facility.

Thus, from a regulatory standpoint, the requirement to use the TID-14844 source term is a licensee commitment (typically expressed as a commitment to Regulatory Guide 1.3 or 1.4) documented in the facility FSAR. The licensee may effect a change in its licensing basis, including the FSAR, by applying for an amendment to its license under §§ 50.90–50.92, or on its own volition within the provisions of § 50.59. Because of the extensive use of the accident source term in the design and operation of a power reactor, and because of the potential impact on postulated accident consequences and margins of safety of a change in such a fundamental design assumption, the NRC concluded that an AST should be implemented by a license amendment under §§ 50.90–50.92.

b. Accident Dose Criteria and Control Room Dose Criteria

The accident dose guidelines for operating reactors licensed before January 10, 1997, are presented in § 100.11. These guidelines are expressed in terms of whole body and thyroid dose. Two guidelines are provided. The first is for the EAB for the 2-hour period immediately following the onset of radioactivity release. The second is for the low population zone (LPZ) for the duration of the event. General Design Criterion 19 (GDC-19), *Control Room*, of Appendix A to 10 CFR Part 50, 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 are expressed in terms of 0.05 Sv (5 rem) whole body dose, or its equivalent to any organ. SRP Section 6.4, *Control Room Habitability Systems,* contains guidance that defines *equivalent* as 0.3 Sv (30 rem) to the thyroid and 0.3 Sv (30 rem) to the skin.

In January 1997, the NRC amended its regulations in 10 CFR Parts 21, 50, 52, 54, and 100 to (1) provide site criteria for future sites and (2) relocate source term and dose requirements for future plants into § 50.34. The guidelines of § 100.11 remain in place as the licensing basis for operating reactors licensed before January 10, 1997. In relocating the source term and dose requirements for future reactors to § 50.34, the NRC retained the requirements for the EAB and the LPZ, but revised the associated numerical dose guidelines 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 guideline for the EAB was expressed in terms of the 2-hour period that yielded the maximum dose. The NRC did not, at that time, amend the control room dose criterion in GDC-19.

In a staff requirements memorandum dated February 12, 1997, the Commission directed that the amended dose guidelines be made applicable to operating plants choosing to use a revised source term. Therefore, an AST cannot be implemented without a modification of the accident dose criteria and the GDC-19 criteria. It is this needed modification that made this rulemaking necessary.

B. Conforming Changes

Part 52 governs the issuance of early site permits, standard design certifications, and combined licenses for nuclear power facilities. Part 52 is used in conjunction with applicable requirements of Part 50. The TMI-related requirements in § 50.34(f) were specifically identified as requirements in § 52.47(a)(1)(ii) to the extent that they are technically applicable. The NRC expects that future plants will use the revised source terms, or an approved alternative, in supporting safety analyses. Because §§ 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) contain specific references to the TID-14844 source term and would otherwise appear to be technically applicable,, these sections needed to be revised in order for new design certifications to use ASTs. In addition, § 52.79(b) would require a combined operating license applicant to use the TID-14844 source term. The control room habitability criteria in GDC-19 were incorporated by reference in § 52.47(a)(i). This criterion is expressed in terms of whole body dose or its equivalent to any part of the body rather than in terms of TEDE. Exemptions from these requirements were necessary for the Westinghouse AP-600 final design approval and design certification. The final rule will address changes to these affected sections in order to avoid the need for exemptions for subsequent applicants under Part 52.

The conforming changes to § 50.34(f) are also applicable to the small subset of specifically listed applicants that had a construction permit application pending on February 16, 1982. The NRC does not expect these applications to be pursued further. However, should one of these applications be reactivated, the applicant will have the option of using an approved alternative to the TID-14844 source term.

II. OBJECTIVE OF FINAL RULE

A. Use of Alternative Source Terms at Operating Reactors

The objective of this final regulatory action is to set up a regulatory framework for the voluntary implementation of ASTs as a change to the design basis at currently licensed power reactors, thereby enabling potential cost-beneficial licensing actions while continuing to maintain existing safety margins and defense in depth.

This is accomplished by the following actions:

- Providing revised accident dose criteria and control room habitability dose criteria that are consistent with the characteristics of the revised source terms and that reflect updated scientific information on radionuclide uptake and metabolism, and also reflect current radiation protection standards; and

- Requiring submittal of a license amendment that contains an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

Because conformance to the final rule is voluntary and will not constitute a backfit, the licensing bases for operating reactors that do not adopt an AST must remain in the regulation. Therefore, the final rule is designated as a new section, § 50.67, applicable to operating reactors licensed before January 10, 1997, that are proposing to use an AST. The existing requirements in Part 100 and GDC-19 are maintained for operating reactors that continue to use the TID-14844 source term.

The NRC has prepared Draft Regulatory Guide DG-1081 and draft SRP Section 15.0.1 in support of this rule. Draft Guide-1081 is being issued for public comment concurrent with the publication of the final rule.

B. Conforming Changes

The objective of this final regulatory action is to eliminate the need for applicants under Part 52 to request exemptions from certain of the NRC's regulations. The need for these exemptions was identified during the Westinghouse AP-600 advanced reactor design certification proceeding.

This is accomplished by the following actions:

- Explicit references to the TID-14844 source term in § 50.34(f) have been revised to read "accident source term." A footnote has been added to define an accident source term in generic terminology (similar language to the corresponding footnote in Part 100).

- GDC-19 has been revised to incorporate a revised dose criterion that is applicable only to applicants for construction permits under this part or for a design certification or combined license under 10 CFR Part 52 who apply on or after January 10, 1997. The current dose criterion remains in effect for those operating reactors that continue to use the TID-14844 source term.

III. ALTERNATIVE APPROACHES

A. Use of Alternative Source Terms at Operating Reactors

The no-action alternative of retaining the existing accident source term was not considered in the development of the final rulemaking activity. In SECY-96-242, the NRC staff made recommendations to the Commission on how the revised source terms could be implemented at operating reactors. In staff requirements memorandum on SECY-96-242, the Commission directed the NRC staff to (1) complete the rebaselining study, (2) complete pilot plant evaluations, (3) commence rulemaking activities, and (4) include the TEDE terminology and the worst 2-hour methodology.

The first alternative considered by the NRC was to continue using current regulations for accident dose criteria and control room dose criteria. This was not considered to be an acceptable alternative. The NRC had previously determined in the January 1997 Part 50 and Part 100 final rule that dose guidelines expressed in terms of whole body and thyroid doses were inconsistent with the use of the revised source terms. With regard to the EAB dose guideline, the NRC also determined that the dose guideline applies to that 2-hour period resulting in the maximum dose.

The second alternative considered by the NRC was to replace the existing guidelines in § 100.11 and the existing criteria in GDC-19 with revised dose criteria. This is not considered to be an acceptable alternative because the provisions of the existing regulations form part of the licensing bases for many of the operating reactors. Therefore, these provisions must remain in effect for operating reactors that do not implement an AST. In addition, this rulemaking alternative would also be inconsistent with the NRC's philosophy of separating plant siting criteria and dose requirements. The approach of establishing the requirements for use of ASTs in a new section to Part 50 while retaining the existing regulations in Part 100 Subpart A and GDC-19 was chosen as the best rulemaking alternative.

The NRC considered alternatives with regard to providing regulatory guidance to support the new section to Part 50. The first alternative was to issue no additional regulatory guidance. This was not considered to be an acceptable alternative because, in the absence of clear regulatory guidance, licensee efforts in preparing applications, and the NRC's review of submitted applications, could be hindered by differences in interpretations and technical positions. This could result in the inefficient use of licensee and NRC resources, could cause licensing delays, and could lead to less uniform and less consistent regulatory implementation. The second alternative was to replace the existing regulatory guides that address accident radiological consequences with new revisions. This was not considered to be an acceptable alternative because the provisions of the existing regulatory guides form part of the licensing bases for many of the operating reactors. Therefore, these provisions must remain in effect for operating reactors that do not implement an AST. The third alternative was to issue a new regulatory guide on the implementation of the revised source terms that would include revised assumptions and acceptable analysis methods for each design basis accident in a series of appendices. The approach of issuing a new regulatory guide was chosen as the best alternative. To provide review guidance for the NRC staff, a new section on design basis radiological analyses using ASTs will be added to the Standard Review Plan.

B. Conforming Changes

Because these revisions are conforming changes for a rule issued earlier, the no-action alternative was not considered to be acceptable. No reasonable alternative was identified for the necessary § 50.34(f) revisions. The reference to TID-14844 needs to be removed.

With regard to a revised control room dose criterion, the revised criterion could have been implemented by changing Part 52 (that cross-references Part 50), by changing § 50.34(a), or by changing GDC-19. A change to GDC-19 was found to be the simplest and clearest approach and, therefore, was considered to be the acceptable alternative.

IV. EVALUATION OF VALUES AND IMPACTS

The NRC has determined that public health and safety and the common defense and security would continue to be adequately protected when the final rule is implemented. The NRC has qualitatively determined that the potential values associated with the revised source terms are substantial enough to justify the rule. This final rule is voluntary for operating reactors. (The conforming changes for Part 52 will be mandatory for future applicants.) The basis for these conclusions is discussed in the sections that follow.

The NRC has not prepared a quantitative value-impact analysis. First, compliance with the rule is voluntary for operating reactors. It is assumed that licensees will pursue implementation of the AST only if they perceive it to be in their interest to do so. Second, it is likely that applications will vary widely with regard to scope and extent, making meaningful quantitative value-impact analyses questionable. In the staff requirements memorandum (SECY-98-289) dated February 25, 1999, the Commission directed the NRC staff to allow licensees the maximum flexibility to pursue technically justifiable applications.

A. Use of Alternative Source Terms at Operating Reactors

1. Values

This rulemaking will allow operating reactors to voluntarily replace the traditional TID-14844 source term with a source term that is based on the insights gained from extensive accident research activities. 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 design of some engineered safety features, such as containment spray systems and containment, ventilation exhaust, and control room charcoal filters, is largely predicated on the radiation doses postulated using these source terms. It is expected that an AST, with its improved understanding of chemical/physical form and release timing, could be used to effect reductions in operational and maintenance requirements associated with some of these systems. These reductions will have economic benefit.

The implementation of an AST does not, in itself, have economic value. It is the modifications to the facility structures, systems, components, and procedures, enabled by an AST, that give rise to the associated values and impacts. Because this is a voluntarily action on the part of the licensee, it is expected that licensees will not pursue applications of an AST unless they perceived it to be in their benefit. Because of this conclusion and the large number of possible applications varying in scope and extent, the NRC has not performed quantitative value-impact analyses. In 1996, NEI informally polled the industry to determine how often and for what uses licensees might apply the NUREG-1465 source terms. Although the poll was conducted informally and does not constitute any commitment to act, the results of the poll indicate the level of interest in the use of an AST. The responses received represented 43 operating power reactors. Of these, 41 reactors plan to use the revised source terms to pursue plant modifications. Anticipated applications include the following:

- change in allowable containment and emergency core cooling system (ECCS) leak rates (24 plants)
- change in isolation valve actuation timing (31 plants)
- simplification of filtration units (27 plants)
- change in mitigation system actuation timing (22 plants)
- change in equipment qualification (2 plants)

The NRC has conducted three public meetings with industry representatives since this rulemaking has been in preparation. These meetings have been well attended by individual utilities, vendors, and owners groups. The NRC has been contacted by a multisite utility expressing an interest in AST applications. The NRC has been informally told by vendor representatives that they are working on applications for several utilities. On the basis of this expression of interest, the NRC concludes that licensees will make extensive use of the AST.

There is an expectation that many of the AST applications may provide concomitant improvements in overall safety and in reduced occupational exposure, as well as economic benefits. Because of the wide range of possible applications and the voluntary nature of this rule, it is not reasonable to quantify possible outcomes. Reductions in occupational exposures may be realized through reductions in maintenance efforts associated with maintaining unnecessarily limiting leakage, timing, or filtration requirements. Improvements in overall safety may be realized through reduced emergency diesel generator loading, improved containment ventilation system performance due to removal of filter media, and closer synchronization of mitigation feature actuation with the onset of major fission product release, to provide just three examples. There may be improvements in safety margins realized due to the upgrading of analysis assumptions, methods, and acceptance criteria.

It is believed that the final rule will result in an improvement in the allocation of resources both for the NRC and for industry. The industry will be allowed to propose applications of ASTs that could reduce unnecessary or ineffective requirements in the facility design basis. Limited resources could be diverted to safety issues of greater significance.

2. Costs

Since the implementation of an AST is a voluntary action on the part of the licensee, licensees are not expected to pursue applications of an AST unless they perceive it to be in their benefit. Because of this conclusion and the large number of possible applications varying in scope and extent, the NRC has not performed quantitative value-impact analyses.

3. Impacts

It is difficult to determine with a degree of accuracy the actual impacts of the final rule since it does not mandate or approve any *specific* source term as a substitute for TID-14844. However, to get some idea of the potential impact, the NRC assumed for purposes of this regulatory analysis that a licensee would seek to replace the traditional TID-14844 source term with a source term that is based on the source terms in NUREG-1465. Using NUREG-1465, the actual accident sequence and progression are not changed; it is the regulatory assumptions regarding the accident that will be affected by substituting the AST. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF) or actual offsite or onsite radiation doses. (Although *actual* doses would not increase, analysis results may show an increase in some *postulated* doses because additional radionuclides will be considered and dose modeling will be more comprehensive.) The accident source terms are used in analyses performed to assess the adequacy of the plant design to contend with a DBA in order to ensure adequate defense in depth and adequate safety margins.

An AST could be used to justify changes in the plant design that could have an impact on CDF or LERF or that could increase offsite or onsite doses. These potential changes are subject to existing requirements in the NRC's regulations. The supporting draft regulatory guide for this rule discusses the need for an evaluation of the impacts of an AST implementation, including consideration of reductions in defense in depth, safety margins, or both. Consistent with Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis*, the draft regulatory guide indicates that probabilistic risk assessment (PRA) insights may have to be considered if the proposed changes to the design basis are not addressed in currently approved NRC staff positions.

The Commission directed the NRC staff to assess the impacts of implementing the NUREG-1465 source terms at operating reactors. The results of this study were presented to the Commission in SECY-98-154, *Results of the Revised (NUREG-1465) Source Term Re-Baselining for Operating Reactors.* The major areas examined were the effect on individual offsite and control room dose, the effect on doses used in equipment environmental qualification, and the effect of potential modifications that might be enabled by the revised source terms. The study also assessed the margin afforded by the revised source terms in comparison to assessments performed using the integrated severe accident assessment code, MELCOR. The study indicated that the impact of implementing the revised source terms at operating reactors will produce lower postulated doses in the majority of cases. The NRC has addressed the exceptions in the draft regulatory guide that is being made available for public comment in conjunction with this final rule. The NRC will also address these exceptions in the processing of the individual license amendments. The MELCOR best-estimate analyses indicated that the design basis dose calculations using the revised source term continue to have a substantial margin (a factor of two or greater). The study also indicated that many of the plant systems that are likely to be considered for modification are not involved in risk-significant sequences and are, therefore, not likely to have a substantial offsite risk impact using a measure such as LERF. At the present time, the only approved alternative to the TID-14844 source term is the source term in NUREG-1465. The NRC expects that any future proposed AST will be subjected to the same level of scrutiny as was used in approving NUREG-1465.

On the basis of these considerations, the NRC concludes that approval of an AST based upon NUREG-1465 will not involve a significant increase in the probability or consequences of accidents previously analyzed, nor will it create a new or different type of accident or result in a significant reduction in safety margin.

The NRC does not intend to approve any source term that is not of the same level of quality as the source terms in NUREG-1465, or that has not had the extensive peer review as did NUREG-1465. The draft regulatory guide contains guidance on acceptable ASTs. Any AST is expected to provide the same level of protection as does the source terms in NUREG-1465. Thus, the NRC concludes that this rule itself is unlikely to have any significant impact on public health and safety and will continue to provide reasonable assurance of adequate protection.

4. Backfit Considerations

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final regulation, and, therefore, a backfit analysis is not required because these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1). The final § 50.67 amends the NRC's regulations by establishing alternate requirements that may be voluntarily adopted by operating reactors licensed before January 1997 that have adopted, or are proposing to adopt, an AST.

5. Impacts on NRC Resources, Other NRC Programs and Other Agencies

The final rule will not affect Federal, State, or local Government agencies, or Agreement State licensees because the rule will affect only the licensing and operation of nuclear power plants that are regulated by the NRC under Part 50. Within the NRC, the responsible office is the Office of Nuclear Reactor Regulation (NRR), which is sponsoring this final rule.

This rule is expected to increase NRC resources needs. Licensees are required by § 50.67 to submit license amendment requests to amend their facility design basis to replace their current accident source term with an AST. These amendment requests will require review by the NRC staff. NRR will bear much of this resource demand. However, it is likely that NRR may request assistance from the Office of Nuclear Regulation Research (RES) in evaluating licensee approaches that differ from the guidance provided in the draft guide. Specialized contractor support could also be needed. Additionally, other NRC offices, for example, the Office of the General Counsel (OGC) and the Office of Administration (ADM), may experience an increase in their workload associated with processing license amendments.

It is not feasible to prepare a detailed quantitative estimate of the potential resource expenditures because of the wide variety expected in the scope and extent of AST applications. The Commission directed the NRC staff to allow the maximum flexibility in pursuing technically justifiable applications, provided that a clear, consistent, and logical design basis was maintained. The NRC staff had proposed a narrower range of potential applications in the rulemaking plan (SECY-98-158, June 30, 1998). As a result of the Commission's direction, the regulatory guidance prepared for this final rule addresses full and selective implementations of an AST. Although the review of a full implementation might be expected to involve more NRC staff resources, these reviews may involve fewer resources in assessing the impact on maintaining a clear, consistent, and logical design basis. Additionally, a licensee proposing a full implementation would need to submit under § 50.67 only once. A selective implementation may represent a greater challenge to the integrity of the facility's design basis, particularly if a licensee pursues multiple selective implementations. The NRC's review of these selective applications will likely require greater diligence. Additionally, licensees would be required to submit a request under § 50.67 for each additional AST application.

The NRC expects that although there may be additional amendment requests related to AST implementation, there may not be a significant increase in the overall license amendment workload or backlog. Licensees tend to prioritize license amendment requests, holding back lower priority requests. To some extent, this is evidenced in the relatively constant year-to-year rate of amendment requests. The NRC does not expect a substantial increase in workload or backlog.

As noted earlier in this analysis, there is industry interest in applications of the AST. In an earlier survey by NEI, 41 reactors polled informally were planning to use the AST to pursue plant modifications. On the basis of discussions with industry personnel and on the expression of interest exhibited in public meetings conducted to date, this estimate (rounded to 40) is deemed to be reasonable.

In reviewing of the Perry pilot project, the NRR technical staff expended approximately 760 hours. The Perry pilot project was the first of its kind. The NRC expects expended staff hours to be less for future amendment requests. For the purpose of this analysis, it will be assumed that 25 percent of the hours expended could be applicable. However, it is also expected that some AST implementations will have a scope or extent that will enable a reduced level of NRC staff review, and that some may require more. To account for this, this analysis will reduce the time estimate by 30 percent. On the basis of these adjustments, the potential NRR resource expenditure is assumed to be about 135 hours or about 0.066 full-time equivalents (FTE). For purposes of comparison, Table 2, *Guidance for Estimating Application Review Hours of Effort*, of NRR Office Letter 803, Revision 2, *License Amendment Review Procedures*, provides an estimate of >120 hours for an amendment of high technical complexity and low similarity between current amendment and precedents. With regard to these reviews, licensees are proposing implementations of an AST only in conjunction with proposed plant modifications requiring review by multiple technical disciplines. In many cases, the licensee is employing new analysis methods not previously considered by the NRC staff. It is these latter two considerations - extent of plant modification and unique analysis methods -- that drive the technical review hour expenditure. Although the Perry pilot project review involved significant support from RES personnel and some contractor support, this level of effort is not expected on each review. To account for the potential need for such support, this analysis will assume 15 percent or 0.01 FTE. Another 5 percent or 0.003 FTE will be assumed for project management, OGC review, and ADM processing, for a total projected expenditure of about 0.08 FTE.

Assuming 40 applications with an average FTE of 0.08 yields 3.2 FTE. The final rule will not be published before the end of FY99. The NRC expects that the pace of submittals of license amendment requests will start relatively slowly and will accelerate as the NRC approves applications and as industry experience increases. On this basis, the NRC expects 10 AST applications (0.8 FTE) in FY2000, 15 AST applications (1.2 FTE) in FY2001 and 15 AST applications (1.2 FTE) in FY2002.

B. Conforming Changes

1. Values

These conforming changes will eliminate the need for future applicants under Part 52 to apply for exemptions from certain paragraphs in § 50.34(f) and GDC-19. This eliminates the costs associated with preparing and processing an exemption request. By eliminating the need for exemptions, the integrity of the regulations will be maintained.

2. Costs

Because the conforming changes will eliminate the need for future applicants under Part 52 to apply for exemptions from certain paragraphs in § 50.34(f) and GDC-19, it is expected that costs will be reduced, not increased.

3. Impacts

Because these are conforming changes for regulations already promulgated, there could be no significant increase in the probability or consequences of accidents previously analyzed, nor would a new or different type of accident be created, nor would there be a significant reduction in safety margins.

The final conforming changes to § 50.34(f) will also be applicable to the small subset of specifically listed applicants that had a construction permit application pending on February 16, 1982. The NRC does not expect these applications to be pursued further. However, if one of these applications would be reactivated, the applicant would be given the option of using an approved alternative to the TID-14844 source term. If an affected applicant chose to use an AST, the impact discussion and conclusions given above for the final § 50.67 would apply.

4. Backfit Considerations

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final regulation and, therefore, a backfit analysis is not required for this final regulation because these amendments would not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1).

- The final changes to § 50.34(f), by removing the explicit reference to TID-14844, allows future applicants under Part 52 to use an AST without the need for seeking exemptions, and allows the small class of applicants for which a construction permit or manufacturing license was pending on February 16, 1982, to use an approved alternative to the TID-14844 source term in showing compliance with § 50.34(f). With the exception of the Westinghouse AP-600 final design approval process, there are no pending Part 52 applications. (Westinghouse requested an exemption from the affected paragraphs in § 50.34(f) to use the revised source term.)

- The final change to GDC-19, requires future applicants under Part 50 or Part 52 after January 10, 1997, to show compliance with the 0.05 Sv (5 rem) TEDE dose criterion. There are no applicants in this status at the present time.

5. Impacts on NRC Resources, Other NRC Programs, and Other Agencies

The final rule will not affect Federal, State, or local Government agencies, or Agreement State licensees, because the rule only affects the licensing and operation of nuclear power plants that are regulated by the NRC under Part 50. Within the NRC, the responsible office is Nuclear Reactor Regulation, which is sponsoring this final rule. No other NRC office is affected by this final rule. These conforming changes will not increase NRC resource needs because the conforming changes will eliminate the need to process exemption requests.

V. DECISION RATIONALE

A. Use of Alternative Source Terms at Operating Reactors

The decision to create a new section in Part 50 (i.e., § 50.67)— and to include the following provisions: the need for a license amendment, the accident dose criteria in § 50.34(a)(1)(ii), and the 0.05 Sv (5 rem) TEDE dose criterion for the control room—was based on the following rationale:

1. The objective of providing a regulatory framework for the voluntary implementation of ASTs as a change to the design basis at currently licensed power reactors. (This would enable potential cost-beneficial licensing actions and would continue to maintain existing safety margins and defense in depth.)

2. The need for accident dose criteria and control room habitability dose criteria that are consistent with the characteristics of the revised source term and that reflect updated scientific information on radionuclide uptake and metabolism, and current radiation protection standards.

3. The provision that an AST be implemented in a facility's design basis by a license amendment, which addresses the NRC concern that the current language of § 50.59 could be interpreted as allowing this change without prior approval. (The NRC has proposed changes to § 50.59. The approach taken in § 50.67 is not inconsistent with the amendments being made to § 50.59.)

4. The results of the NRC rebaselining study that did not identify any significant concerns related to implementation of the revised source term.

5. The NRC philosophy of separating plant siting from plant design, as evidenced by the January 1997 Part 50 and Part 100 final rule.

6. The need to maintain the existing licensing basis for the operating reactors that continue to use the TID-14844 source term.

B. Conforming Changes

The decision to address needed conforming changes to Part 50 and to include the 0.05 Sv (5 rem) TEDE dose criterion for the control room was based on the following rationale:

1. The desire to eliminate the need for exemptions from compliance with the affected sections.

2. The need for control room habitability dose criteria that are consistent with the characteristics of the revised source term and that reflect updated scientific information on radionuclide uptake and metabolism, and current radiation protection standards.

VI. IMPLEMENTATION

In the interest of facilitating stakeholder participation in this rule and allowing interested licensees to proceed with the development of applications, the Commission decided to separate development of the proposed rule from the proposed draft regulatory guide and draft SRP section. This regulatory analysis addresses the final rule, Draft Regulatory Guide DG-1081, and draft SRP Section 15.0.1.

Because this is a voluntary rule for operating reactors, there will be no required schedule for implementation on the part of licensees. The final rule will be made effective 30 days following publication. No backfit will be involved.

The NRC expects to finalize the regulatory guide and the SRP section in April 2000. The NRC expects to use the guidance in DG-1081 and the draft SRP Section 15.0.1 in reviewing any license amendment requests received between the effective date of the final rule and the finalization of the regulatory guide and SRP section.

The final rule language is provided in the *Federal Register* notice for which this regulatory analysis applies. The accident dose criteria and the control room dose criteria in the final rule are readily quantifiable and enforceable. These guidelines and criteria are performance based (i.e., the final rule does not prescribe how to meet the requirement).

VII. REFERENCES

1. *Use of Alternative Source Terms at Operating Reactors*, Proposed Rule, 64 FR 12117, March 11, 1999

2. *Draft Regulatory Analysis: Revision of 10 CFR Parts 21, 50, and 54*, Contained in SECY-98-289, December 15, 1998

3. *Accident Source Terms for Light-Water Nuclear Power Plants*, NUREG-1465, February 1995

4. *Calculation of Distance Factors for Power and Test Reactor Sites*, Technical Information Document (TID) 14844, March 1962

5. *Results of the Revised (NUREG-1465) Source Term ReBaselining for Operating Reactors*, SECY-98-154, June 1998

6. *Amendments to 10 CFR Parts 50, 52, and 100, and Issuance of New Appendix S to Part 50,* SECY-96-118, May 1996

7. *Use of the NUREG-1465 Source Term at Operating Reactors*, SECY-96-242, November 1996

Attachment 1

ADAPTATION OF NUREG-1465 SOURCE TERMS FOR EXTENDED BURNUP FUEL

PURPOSE

This attachment addresses technical issues related to the derivation of the source term identified in Draft Regulatory Guide DG-1081 with regard to extended burnup fuel. The source term presented in DG-1081 is specified as being applicable for use with burnups no greater than 62 GWD/MTU averaged over the length of the peak rod.

DISCUSSION

In NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants*, the NRC provides a more realistic estimate of the release of fission products from the fuel into the containment in terms of timing, radionuclide composition and magnitude, quantities, and physical and chemical form. The intent of NUREG-1465 was to capture the major relevant insights available from recent severe accident research on the phenomenology of fission product release and transport behavior. The approach taken in NUREG-1465 presents, for regulatory purposes, a more realistic portrayal of the amount of fission products present in the containment from a postulated severe accident.

The more significant aspects of the NUREG-1465 document that have a bearing on its use by currently operating power reactors are the following:

1. The accident source terms defined in NUREG-1465 were derived from examination of a set of severe accident sequences for light water reactors (LWRs) of current design. NUREG-1465 briefly addresses other design basis analyses, including a fuel handling accident (FHA) and reactivity insertion accidents (RIAs). However, the overall focus in developing NUREG-1465 was largely on loss of coolant accidents (LOCAs). Although NUREG-1465 addresses iodine species in the containment atmosphere, it is silent with regard to the iodine species in the reactor coolant system, in containment sump water, in recirculated fluids, or other process streams.

2. NUREG-1465 contains a caveat indicating that the source term may not be adequate for fuel irradiated above 40 GWD/MTU. The document reports on preliminary research results indicating that fuel irradiated at levels in excess of about 40 GWD/MTU may be more prone to failure during design basis RIAs. This is a significant limitation because the NRC has been allowing burnups up to 60 GWD/MTU, averaged over the length of the peak rod.

3. NUREG-1465 identified gap fractions of 3 percent for events with long-term cooling. These data are inconsistent with some published data for extended burnup fuels.

4. NUREG-1465 identifies five release phases addressing release situations that exceed those typically postulated in design basis analyses.

As a result of these considerations, the NRC decided not to simply endorse NUREG-1465 in its entirety, but to establish, in the draft regulatory guide, an acceptable source term for design basis accidents (DBAs) at currently operating power reactors with fuel burnups no greater than 62 GWD/MTU. The remainder of this appendix discusses the rationale behind the source terms provided in DG-1081.

ANALYSIS

1. Applicability to Other Accidents

The NRC decided to address, in the draft regulatory guide, all DBAs that result in fuel damage. Because some facility analyses postulate exceeding departure from nucleate boiling (DNB) and, hence, fuel damage, for events such as steam generator tube ruptures or main steam line breaks, these events are addressed by the draft guide. The draft guide allows licensees to propose, for NRC consideration, other fuel damage estimate methods for the purpose of establishing radioactivity releases. The NRC expects that these methodologies may show no fuel damage in these secondary events. The NRC deemed it appropriate to address these lesser accidents in the interest of co-locating all applicable guidance related to use of ASTs in a single regulatory guide.

The iodine species identified in NUREG-1465 are specified for releases to containment. The draft guide addresses the following other pathways:

a. For the DBA LOCA containment sump activity, the draft guide directs that, except for noble gases, all fission products released from the fuel should be assumed to be instantaneously mixed in the sump water at the time of release from the core. This is consistent with the traditional treatment of this pathway provided in SRP Section 15.6.5. However, the draft guide does allow the use of suitable conservative mechanistic transport models to evaluate the transport of containment airborne activity to the sump water. NUREG-1465 did not address the iodine species available for release from ECCS leakage during a DBA LOCA. The NRC decided to specify that the release to the atmosphere from the flashing or evaporation of this liquid was elemental iodine. The NRC also assumes that 3 percent of the released iodine is converted to organic iodine by reaction with organic materials in the buildings that enclose these systems. The 3 percent conversion was assumed by the authors of NUREG-1465. This results in an iodine species breakdown of 97 percent elemental and 3 percent organic. Although there are insufficient data to establish this breakdown with a high degree of certainty, the NRC believes the specified breakdown to be adequately conservative. The NRC assumes that the aerosol and particulate forms will be largely retained in the liquid phase.

b. For the FHA, the NRC has opted to retain the iodine species currently specified in Regulatory Guide 1.25 as 99.75 percent elemental and 0.25 percent organic. Since the release of the elemental species is mitigated by pool scavenging whereas the organic forms are not, the species released to the atmosphere would be 44 percent elemental and 55 percent organic. This breakdown is based on a pool decontamination factor (DF) of 500 for elemental forms and a DF of 1 for organic forms. The data in Regulatory

Guide 1.25 are based on an earlier NRC report (G. Burley, 1971). That report assumed the release to be elemental with some conversion to organic forms based on an assumption of 0.5 ppm methane created from trace amounts of impurities in the uranium oxide used in the fuel. Although the NRC believes the assumption that the release is largely elemental is likely very conservative, the NRC has no basis at this time to revise this traditional assumption. However, the NRC believes that the implied conservatism provides adequate margin to support the NRC's decision to increase the allowable pool DF credit.

c. For the BWR rod drop accident, PWR main steam line break, PWR steam generator tube rupture, PWR locked rotor accident, and the PWR rod ejection accident, the NRC has specified an iodine species of 97 percent elemental and 3 percent organic forms for releases to the environment via steam generators or main condensers. This species breakdown applies to the activity released from the component and not the activity contained in the liquid in the component. The NRC assumes significant retention of aerosol and particulate forms. The bases for the numeric breakdown are given above. Although there are insufficient data to establish this breakdown with a high degree of certainty, the NRC believes that the specified breakdown is adequately conservative.

d. For the BWR main steam line break and the in-containment release components of the PWR locked rotor and rod ejection accidents, the NRC has specified that releases due to fuel damage assume the 95 percent, 4.85 percent elemental, and 0.15 percent organic iodine species breakdown provided in NUREG-1465. The NRC believes that these pathways fall within the scope of the NUREG-1465 species data.

2. <u>Applicability of NUREG-1465 Data to Extended Burnup Fuel</u>

The NRC has developed an agency program plan for high-burnup fuel. This plan was sent to the Commission by memorandum dated July 6, 1998. This plan addresses nine issues related to utilization of fuel up to the current limit of 62 GWD/MTU. Issue 7 addresses source term and core-melt progression. The discussion of this issue is summarized below.

The NUREG-1465 source term may not be applicable for fuel irradiated to high burnup levels (in excess of about 40 GWD/MTU). It is generally known that at higher burnup levels the gap inventory will increase, fuel particle behavior will be different, and the isotopics will shift. The main effects that might impact source terms at high burnup levels are (a) embrittlement of the fuel cladding, (b) an increase in the release of fission gases from fuel pellets during normal operation, (c) fragmentation of fuel pellets, and (d) a shift in the spectrum of the fission products produced as plutonium fission becomes more significant.

a. The increased fuel cladding embrittlement is not expected to significantly affect the outcome of uninterrupted core-melt accidents. This conclusion is based on the fact that the DBA LOCA ultimately releases a significant fraction of the core inventory of fission products, including all of the activity in the gap. For non-LOCA events, clad embrittlement could lead to increased cladding failure and increased radioactivity releases. However, existing fuel design regulatory limits provide reasonable assurance that embrittlement will not have a significant impact on postulated releases. The draft regulatory guide advocates the use of fuel damage estimation methods based on integrated enthalpy deposition. The criterion against which the estimates are compared can be adjusted for embrittlement. For FHAs, the NRC's traditional assumption that all rods in the dropped bundle are failed provides margin. However, the NRC has allowed some licensees to base these fuel damage estimates on more realistic stress analyses. It is expected that these licensees would address potential embrittlement effects in performing these estimates.

b. For DBA LOCAs, gap activity represents only a small part of the source term so that even large changes in gap activity would not have a large impact on the assessed accident consequences. However, for accidents involving only gap activity, for example, fuel handling accidents, the change in gap activity could be significant. NUREG/CR-5009, *Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors*, evaluated the environmental effects of the use of extended burnup fuel and concluded that no significant adverse effects would occur as long as the peak rod average burnup was no greater than 60 GWD/MTU. This report evaluated the release fractions for several nuclides at this burnup level. These projected gap fractions (i.e., fraction of core inventory located in gap) are shown below:

Kr-85 and stable noble gases	0.14
Kr-87	0.007
Kr-88	0.01
Xe-133	0.05
Xe-135	0.02
I-131	0.12
Cs-134	0.11
Cs-137	0.17

The NRC has performed calculations using the FRAPCON-3 code, which has been validated out to 65 GWD/MTU. These calculations showed a gap release of 11 percent for long-lived noble gases. The gap release for I-131 would be approximately 9.4 percent. These data were for 15 x 15 PWR fuel with a peak rod average burnup of 65 GWD/MTU. These data were presented to the Advisory Committee on Reactor Safeguards (ACRS) on March 11, 1999. The FRAPCON-3 results confirm the applicability of the NUREG/CR-5009 data out to 65 GWD/MTU.

On this basis, the NRC has decided to specify the following gap releases for non-LOCA events for fuel up to the current licensed limit of 62 GWD/MTU:

I-131	0.12
Kr-85	0.15
Other noble gases	0.10
Other Halogens	0.10
Alkali Metals	0.10

The NRC has hired Pacific Northwest National Laboratory (PNNL) to update NUREG/CR-5009 for fuel burnups to 62 GWD/MTU and higher. This effort is expected to result in a change in the assumed gap fractions for fuel handling accidents. These data will not be available until after the draft guide is published for comment. The agency program on high burnup fuel is continuing. Should data become available that indicate the NRC position to be non-conservative, these fractions will be revised in the final regulatory guide. However, on the basis of data currently available, the NRC believes that specified gap fractions are suitable for use in DBA analyses.

c. Although fuel fragmentation has been observed at high burnup levels, it appears fragments are dispersed by washout and there may be no means to get that material into the atmosphere as aerosol particles. In contrast, particulate releases included in the source term are lifted from the core as high temperature gases that condense as aerosol particles. Thus, fuel fragmentation is not expected to increase the consequences of a core melt.

d. The draft guide provides for the core inventory to be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and rated thermal power (including applicable penalty). Therefore, issues related to the increased core inventory and isotopic spectrum shifting will be addressed directly by each applicant.

As noted above, experimental results indicate that the NRC's current methods for evaluating reactivity insertion accidents, for example, rod drop and rod ejection accidents, may underestimate the activity released from the fuel. The NRC's current analysis methodology assumes a localized thermal condition that results in failure of the fuel cladding and the release of activity in the fuel pellets that have exceeded melt temperatures. The gap activity assumption is not affected because the NRC assumes the instantaneous release of the gap fraction that was generated preceding the event. Thus, the gap fractions tabulated above and in the draft guide are appropriate.

The NRC's assumption regarding the melt release may underestimate the release as recent test results indicate a significant activity release from the pellet without exceeding melt temperature. However, the NRC has concluded that there is no reason to change currently approved burnup levels unless the confirmatory research program demonstrates the need for change. The NRC came to this conclusion on the basis of (1) probability of these accidents is low and (2) generic plant transient analyses indicate that the energy inputs during these events are low and will remain below the relevant test data failure levels. The NRC is participating in new international programs that will reassess present conclusions in 3—5 years when significant new data become available.

Attachment 5

Environmental Assessment

And

Finding

Of

No Significant Environmental Impact

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT ENVIRONMENTAL IMPACT

REVISION OF 10 CFR PARTS 21, 50, AND 54

The Nuclear Regulatory Commission (NRC) is amending its regulations to allow the holders of operating licenses at currently operating reactors to voluntarily amend their design bases to replace the current accident source term with an alternative source term such as those described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."

The proposed rule was published for public comment, and availability of the draft environmental assessment was noticed on March 11, 1999 (64 FR 12117). No comments on the draft environmental assessment were received. Therefore, no substantive changes have been made in this environmental assessment.

Identification of Action

The NRC is amending 10 CFR Part 50 by adding a new section, § 50.67, to address the use of alternative accident source terms. Section 50.67 would apply to all holders of operating licenses issued before January 10, 1997, and holders of renewed licenses under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, that seek to amend their facility design bases to replace the current accident source term with an alternative source term on or after the publication date of the final regulation. These licensees are required by § 50.67 to evaluate the radiological consequences of the design basis accidents previously analyzed in the safety analysis report and to request a license amendment under § 50.90. Acceptance criteria for the accident radiological consequence analyses appear in § 50.67. These criteria consist of accident dose guidelines for evaluating of releases of radioactivity to the environment and the resulting exposures to persons off site, and of dose criteria for plant personnel occupying the control room during postulated accidents.

The final rule amends a current regulation by establishing alternate requirements that licensees may voluntarily adopt. The NRC has determined that the existing analytical approach based on the current source term continues to be adequate to protect public health and safety; therefore, the NRC does not intend to backfit the alternative source terms or the changes in accident dose guidelines and control room habitability criteria for operating power reactors. Because the final revision to the regulation does not constitute a backfit, the bases for existing nuclear power plants must be preserved, and the current accident dose guidelines in § 100.11 and the current control room habitability criteria of Appendix A to 10 CFR Part 50 will remain in effect for licensees that do not apply to use an alternative source term.

The NRC is also amending 10 CFR Part 50 by revising 10 CFR Part 50, Appendix A, General Design Criterion (GDC) -19, to allow use of a dose criterion based on total effective dose equivalent. The revised criterion, which is an alternative to the current dose criterion in GDC-19, may be used only by applicants for construction permits under Part 50, applicants for design certification or combined licenses under 10 CFR Part 52, who apply on or after January 10, 1997, and holders of operating licenses using an alternative source term.

Need for the Action

Use of Alternative Source Terms

Current operating light-water reactors were licensed, in part, on the basis of safety analyses that used the fission product release assumptions of Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (1962). Although initially used in the evaluating proposed reactor sites, these fission product release assumptions, known collectively as the "source term," have been used in several regulatory applications related to light-water reactors. This source term was a key input to many of the design analyses associated with currently operating reactors and figures significantly in the design bases for these facilities. Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, physical form, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," which utilized these source term insights to produce revised estimates of the accident source terms. These source terms are described in terms of radionuclide composition and magnitude, physical and chemical form, and timing of release. For design basis accident assessments, the NUREG-1465 source terms are comparable to the TID-14844 source term with regard to the magnitude of the noble gas and radioiodine release fractions. However, the alternative source terms provide a more representative description of the radionuclide composition and release timing.

NUREG-1465 gave alternative 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 terms. These source terms were derived by examining of 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 applicable to evolutionary and passive LWR designs. The NRC considered the applicability of the alternative source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term still adequately protected public health and safety and that operating reactors licensed under this approach would not be required to reanalyze design basis accidents using the alternative source terms. The NRC also concluded that some licensees might wish to use alternative source terms in analyses to support operational flexibility and cost-beneficial licensing actions to reduce unnecessary regulatory burden.

In January 1997, the NRC amended its regulations in 10 CFR Parts 21, 50, 52, 54, and 100 (61 FR 65157). This regulatory action provided siting criteria for future sites and relocated source term requirements for future plants to Part 50. Since the dose requirements tend to affect reactor design rather than siting, they were also relocated to Part 50. 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 of Part 100, respectively. The revised reactor site criteria were added as Subpart B of Part 100, and revised source term and dose requirements were relocated to § 50.34. The existing source term and dose requirements of Subpart A of Part 100 would remain as the licensing bases for operating reactors that do not elect to use an alternative source term. The NRC retained the requirements for the exclusion area and the low population zone, but revised the associated numerical dose guidelines to replace the doses for the whole body and the thyroid gland with a single total effective dose equivalent (TEDE) value.

The dose guidelines for the whole body and thyroid and for the immediate 2-hour exposure period, were largely predicated by the assumed source term being predominantly noble gases and radioiodines instantaneously released to the containment and the assumed "single critical organ" method of modeling the internal dose used when Part 100 was originally published. However, the current dose guidelines, focusing on doses to the thyroid and whole body, assume that the major contributor to doses is radioiodine. Although this assumption may be appropriate with the TID-14844 source term, 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 forms of radioiodine in the alternative source terms are more amenable to mitigation and, as such, radioiodine may not always be the predominant radionuclide in an accident release. The alternative source terms include more radionuclides than did the TID-14844 source term, as implemented in regulatory guidance. The whole body and thyroid dose guidelines ignore these contributors to dose. 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 alternative source terms postulate that the core inventory is released in phases over several hours, with the most significant release beginning at about 30 minutes after 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 alternative source terms. A detailed rationale for the use of 0.25 Sv (25 rem) TEDE as an accident dose guideline and the use of the 2-hour exposure period resulting in the maximum dose for future LWRs is provided at 61 FR 65157. This rationale also applies to operating reactors that elect to use the alternative source term. The NRC considers that it is technically appropriate and logical to extend the dose guidelines, established for future LWRs using the alternative source term, to operating reactors that elect to use the same alternative source term.

The NRC determined that accident dose guidelines and control room habitability criteria used with the alternative source terms should be expressed in terms of TEDE, and that the 2-hour exposure period should be based on the 2-hour period that yields the maximum dose. The final § 50.67 incorporates these acceptance criteria.

Conforming Change to GDC-19

The revision to GDC-19 is not related to the use of alternative source terms at operating reactors but corrects 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, would 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 recent Part 100 rulemaking (61 FR 65157) established accident TEDE guidelines (in § 50.34) for applicants under Part 52, but did not establish a alternative control room dose criterion. Therefore, exemptions to the dose criterion in the current GDC-19 were necessary in the design certification process for the Westinghouse AP-600 advanced light water reactor in order to allow the use of the 0.05 Sv

(5 rem) TEDE criterion deemed necessary for use with the alternative source terms. The revision eliminates the need for exemptions by future applicants under Part 52. This change will also apply to future applications under Part 50 that are filed on or after January 10, 1997.

Environmental Impacts of the Action

The implementation of an alternative source term at an operating power reactor would replace the traditional TID-14844 source term with a source term that is based on the insights gained from extensive accident research activities. Only the regulatory assumptions regarding the accident would be affected by substituting an alternative source term. The actual accident sequence and progression are not changed. By itself, use of an alternative source term would not increase the core damage frequency (CDF) or the large early release frequency (LERF) or actual offsite or onsite radiation doses. (Although *actual* doses would not increase, analysis results might show an increase in some *postulated* doses because additional radionuclides would be considered and dose modeling would be more comprehensive.) The source term is used to analyze the adequacy of the plant design to contend with a design basis accident (DBA) in order to ensure adequate defense in depth and adequate safety margins. The alternative source term could be used to justify changes in the plant design that could affect the CDF or the LERF, increase offsite or onsite doses, or other environmental impacts. 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 Commission directed the NRC staff to assess the effects of implementing the alternative source term at operating reactors. The results of this study were presented to the Commission in SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Re-Baselining for Operating Reactors." The major effects examined were the effect on individual offsite and control room doses, the effect on doses used in equipment environmental qualification, and the effect of modifications that might be allowed by the alternative source term. The study also assessed the margin afforded by the alternative source term in comparison to assessments performed using the integrated severe accident assessment code, MELCOR. The study indicated that implementing the alternative source term at operating reactors would give lower postulated doses in most cases. The NRC has addressed the exceptions in the Draft Guide-1081, that is being published for comment in conjunction with this final rule.

The NRC will also address these exceptions in processing the individual license amendments. The best-estimate MELCOR analyses indicated that the design basis dose calculations using the alternative source terms still have a substantial margin (a factor of 2 or greater). The study also indicated that many of the plant systems that are likely to be considered for modification are not involved in risk-significant sequences and are, therefore, not likely to have a substantial offsite risk impact, using a measure such as the LERF.

There is an expectation that many of the alternative source term applications may improve safety, reduce occupational exposure, and save money. In light of the wide range of possible applications and the voluntary nature of this final rule, it is not feasible to quantify possible outcomes. Occupational exposures may be reduced through reduced maintenance associated with maintaining unnecessarily limiting leakage, timing, or filtration requirements. Overall safety may be improved through (1) staged or reduced emergency diesel generator loading, (2) improved containment ventilation system performance due to removal of filter media, and (3) closer synchronization of accident mitigation feature actuation with the onset of major fission product releases. Safety margins may be increased by the more realistic analysis assumptions, methods, and acceptance criteria.

Based on the conclusions of the re-baselining study, the radiological consequences of DBAs are not increased by the use of the alternative source term. The final dose guidelines are comparable, in level of protection, to the existing guidelines. The final rule does not affect nonradiological plant effluents and has no other environmental impact. Therefore, the NRC concludes that there are no significant nonradiological environmental impacts associated with the amendments to the regulations.

Alternatives to the Action

As required by Section 102(2)(E) of National Environmental Policy Act (NEPA) (42 U.S.C.A. 4332(2)(E)), the NRC staff considered possible alternatives to the final action. Most of the alternatives considered involved administrative details such as location of the final rule and the means of providing regulatory guidance. These alternatives are neutral with regard to environmental impact and will not be considered further. With regard to environmental impacts, the alternatives are (1) to retain the existing accident source term (the no-action alternative) or (2) to allow the use of the alternative source term.

Retaining the existing accident source term was considered unacceptable because it would preclude the potential reductions in unnecessary regulatory burden, potential improvements in overall safety, and potential reductions in occupational exposure. The environmental impact of a postulated DBA would be unchanged. The foreclosure of potential improvements in safety and reductions in occupational exposure could prevent some actions that could reduce the risk and/or consequences of accidents. Because it is not possible to predict with any degree of certainty the source term applications that licensees may voluntarily propose, these applications were not evaluated further.

The second alternative, allowing the voluntary use of the alternative source term at operating plants, including the use of dose guidelines and dose criteria consistent with the alternative source term, would establish the requirements for use of an alternative source term in a new section to Part 50 while retaining the existing regulations in 10 CFR will 100 Subpart A and GDC-19. This was chosen as the better approach. The final rule would improve the allocation of NRC and for industry resources. They could propose applications of an alternative source term that could reduce unnecessary or ineffective requirements in the facility design basis. The NRC and the industry would gain from having appropriate regulatory requirements and guidance to facilitate preparation and NRC staff review of licensee submittals. Resources could be diverted to safety issues of greater significance. The environmental impacts of the proposed use of the alternative source term were addressed earlier in this assessment, and it was concluded that there would be no significant environmental impact. Because of the potential safety and economic benefits, this alternative is clearly superior to the no-action alternative.

Alternative Use of Resources

No alternative use of resources was considered. The final rule applies only to existing operating reactors and the use of an alternative source term for analysis purposes has no impact on the use of resources. Although this rule also makes conforming changes related to future plant licensing, the environmental impact of the future plant licensing would, by regulation, be assessed as part of the plant licensing.

Agencies and Persons Consulted

The NRC staff developed the final rule and this environmental assessment. No outside agencies or consultants were used in developing this assessment. The NRC staff obtained advice from the NRC Advisory Committee on Reactor Safeguards.

The NRC published the proposed rule for public comment on March 11, 1999 (64 FR 12117). The NRC summarized the draft environmental assessment in the statements of consideration for the proposed rule and noticed the availability of the full document. The NRC requested comments on any aspect of the environmental assessment. No comments were received on the environmental assessment. Accordingly, no substantive changes have been made in this assessment.

Copies of the Federal Register notice for the proposed rule and draft environmental assessment were distributed to each State Liaison Officers with a request for comments. No comments were received.

Finding of No Significant Environmental Impact

The final amendments to 10 CFR Parts 21, 50, and 54 to allow the holders of operating licenses at currently operating reactors to voluntarily amend their design bases to replace the current accident source term with an alternative source term, do not have a significant effect on the quality of the human environment.

This conclusion is based foregoing environmental assessment and on the following:

1. The alternative accident source term and the accident dose guidelines were incorporated into the NRC's regulations in Parts 50 and 100 for future plant licensing by a final rulemaking on January 10, 1997. The environmental assessment for that final rule made a finding of no significant impact. Because this final rule would be a logical extension of those provisions to operating reactors, a similar finding is appropriate.

2. The alternative source term reflects the significant advances that have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. This alternative source term provides more physically based estimates of the accident source term. The NRC sponsored

significant review efforts by peer reviewers, foreign research partners, industry groups, and the general public (57 FR 33374).

References

1. "Use of Alternative Source Terms at Operating Reactors," Proposed Rule, 64 FR 12117, March 11, 1999.

2. "Draft Environmental Assessment: Revision of 10 CFR Parts 21, 50, and 54," Attached to SECY-98-289, December 15, 1998.

3. "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

4. "Calculation of Distance Factors for Power and Test Reactor Sites," Technical Information Document (TID) 14844, March 1962.

5. "Results of the Revised (NUREG-1465) Source Term Re-Baselining for Operating Reactors," SECY-98-154, June 1998.

6. "Amendments to 10 CFR Parts 50, 52, and 100, and Issuance of New Appendix S to Part 50," SECY-96-118, May 1996.

Attachment 6

Draft

Congressional Letters

and

Congressional Review Act Forms

The Honorable Joe L. Barton, Chairman
Subcommittee on Energy and Power
Committee on Commerce
United States House of Representatives
Washington, DC 20515

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee are copies of a public announcement and a final amendment to 10 CFR Parts 21, 50, and 54. The final rule will allow holders of operating licenses at currently operating reactors to voluntarily amend their design bases to replace the current accident source term used for accident radiological analyses with an alternative source term such as those described in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants.* The NRC is also making some changes to various sections of its regulations to conform with revisions implemented earlier. The proposed rule was published for public comment in the *Federal Register* on March 11, 1999.

In addition to the final rule, the NRC is also announcing the availability for public comment of a draft regulatory guide, DG-1081, titled *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* and a draft Standard Review Plan (SRP) section, 15.0.1, titled *Radiological Consequence Analyses Using Alternative Source Terms.* These proposed documents will provide regulatory guidance that supports the final rule.

Since the publication of the current accident source term in 1962, 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 effort started by the NRC and the industry after the accident at Three Mile Island (TMI). The final rule will enable currently licensed power reactors to propose applications of an alternative source term that could reduce unnecessary or ineffective requirements in the facility design basis, thereby reducing the regulatory burden. It is believed that this rule will also result in an improvement in the allocation of resources both for the NRC and for industry. Also, there is an expectation that many of the alternative source term applications may provide concomitant improvements in overall safety and in reduced occupational exposure, as well as economic benefits.

The NRC staff has determined that the public health and safety and the common defense and security will continue to be adequately protected after the rule is implemented.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure: Public Announcement
Federal Register Notice

cc: Representative Ralph M. Hall

The Honorable James N. Inhofe, Chairman
Subcommittee on Clean Air, Wetlands, Private
Property and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee are copies of a public announcement and a final amendment to 10 CFR Parts 21, 50, and 54. The final rule will allow holders of operating licenses at currently operating reactors to voluntarily amend their design bases to replace the current accident source term used for accident radiological analyses with an alternative source term such as those described in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants*. The NRC is also making some changes to various sections of its regulations to conform with revisions implemented earlier. The proposed rule was published for public comment in the Federal Register on March 11, 1999.

In addition to the final rule, the NRC is also announcing the availability for public comment of a draft regulatory guide, DG-1081, titled *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* and a draft Standard Review Plan (SRP) section, 15.0.1, titled *Radiological Consequence Analyses Using Alternative Source Terms.* These proposed documents will provide regulatory guidance that supports the final rule.

Since the publication of the current accident source term in 1962, 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 effort started by the NRC and the industry after the accident at Three Mile Island (TMI). The final rule will enable currently licensed power reactors to propose applications of an alternative source term that could reduce unnecessary or ineffective requirements in the facility design basis, thereby reducing the regulatory burden. It is believed that this rule will also result in an improvement in the allocation of resources both for the NRC and for industry. Also, there is an expectation that many of the alternative source term applications may provide concomitant improvements in overall safety and in reduced occupational exposure, as well as economic benefits.

The NRC staff has determined that the public health and safety and the common defense and security would continue to be adequately protected after the rule is implemented.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure: Public Announcement
Federal Register Notice

cc: Senator Bob Graham

24722

Submission of Federal Rules Under the Congressional Review Act

☑ **President of the Senate** ☐ **Speaker of the House of Representatives** ☐ **GAO**

Please fill the circles electronically or with black pen or #2 pencil.

1. Name of Department or Agency	2. Subdivision or Office
Nuclear Regulatory Commission	Office of Congressional Affairs

3. Rule Title
Final Rule: Use of Alternative Source Terms at Operating Reactors, amendments to 10 CFR Part 21, 50, and 54

4. Regulation Identifier Number (RIN) or Other Unique Identifier (if applicable)
3150 AG12

5. Major Rule ○ Non-major Rule ●

6. Final Rule ● Other ○ ____________

7. With respect to this rule, did your agency solict public comments? Yes ● No ○ N/A ○

8. Priority of Regulation (fill in one)

● Economically Significant; or
Significant; or
Substantive, Nonsignificant

○ Routine and Frequent or
Informational/Administrative/Other
(Do not complete the other side of this form if filled in above.)

9. Effective Date (if applicable) 30 days following notice in Federal Register

10. Concise Summary of Rule (fill in one or both) attached ● stated in rule ○

Submitted by: ____________________ (signature)

Name: Dennis K. Rathburn

Title: Director, Office of Congressional Affairs

For Congressional Use Only:

Date Received: ____________

Committee of Jurisdiction: ____________________

3/23/99

	Yes	No	N/A
A. With respect to this rule, did your agency prepare an analysis of costs and benefits?	●	○	○
B. With respect to this rule, by the final rulemaking stage, did your agency			
1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
2. prepare a final Regulatory Flexibility Analysis under 5 U.S.C. § 604(a)?	○	●	○
C. With respect to this rule, did your agency prepare a written statement under § 202 of the Unfunded Mandates Reform Act of 1995?	○	●	○
D. With respect to this rule, did your agency prepare an Environmental Assessment or an Environmental Impact Statement under the National Environmental Policy Act (NEPA)?	●	○	○
E. Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	●	○	○
F. Did you discuss any of the following in the preamble to the rule?	●	○	○
• E.O. 12612, Federalism	○	●	○
• E.O. 12630, Government Actions and Interference with Constitutionally Protected Property Rights	○	●	○
• E.O. 12866, Regulatory Planning and Review	○	●	○
• E.O. 12875, Enhancing the Intergovernmental Partnership	○	●	○
• E.O. 12988, Civil Justice Reform	○	●	○
• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	●	○

• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify)

E.O. 12898

AGENCY:	Nuclear Regulatory Commission
TITLE OF ACTION:	Use of Alternative Source Terms at Operating Reactors
LEVEL OF SIGNIFICANCE:	Non-major rule
UPCOMING ACTION:	Final rule
AGENCY IDENTIFICATION:	RIN 3150-AG12
ESTIMATED DATE OF ISSUANCE:	October 1999
STATUTORY OR JUDICAL DEADLINE:	N/A

DESCRIPTION Of ACTION:

The final rule will amend the Commission's 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 a revised source term. This revised source term was developed from the results of a major research effort to obtain a better understanding of fission-product transport and release mechanisms in light-water-reactors under severe accident conditions. This action will allow interested licensees to pursue cost beneficial licensing actions to reduce regulatory burden without comprising the margin of safety of the facility. The Commission is also amending its regulations to revise certain sections to conform with final rule making published on January 10, 1997 (61 FR 65157).

24722

Submission of Federal Rules Under the Congressional Review Act

☐ President of the Senate ☑ Speaker of the House of Representatives ☐ GAO

Please fill the circles electronically or with black pen or #2 pencil.

1. Name of Department or Agency Nuclear Regulatory Commission	2. Subdivision or Office Office of Congressional Affairs

3. Rule Title
Final Rule: Use of Alternative Source Terms at Operating Reactors, amendments to 10 CFR Part 21, 50, and 54

4. Regulation Identifier Number (RIN) or Other Unique Identifier (if applicable)
3150 AG12

5. Major Rule ○ Non-major Rule ●

6. Final Rule ● Other ○ ____________________

7. With respect to this rule, did your agency solicit public comments? Yes ● No ○ N/A ○

8. Priority of Regulation (fill in one)

● Economically Significant; or Significant; or Substantive, Nonsignificant

○ Routine and Frequent or Informational/Administrative/Other (Do not complete the other side of this form if filled in above.)

9. Effective Date (if applicable) 30 days following notice in Federal Register

10. Concise Summary of Rule (fill in one or both) attached ● stated in rule ○

Submitted by: ____________________ (signature)

Name: Dennis K. Rathburn

Title: Director, Office of Congressional Affairs

For Congressional Use Only:

Date Received: ____________

Committee of Jurisdiction: ____________________

3/23/99

		Yes	No	N/A
A.	With respect to this rule, did your agency prepare an analysis of costs and benefits?	●	○	○
B.	With respect to this rule, by the final rulemaking stage, did your agency			
	1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
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E.	Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	●	○	○
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	• E.O. 12630, Government Actions and Interference with Constitutionally Protected Property Rights	○	●	○
	• E.O. 12866, Regulatory Planning and Review	○	●	○
	• E.O. 12875, Enhancing the Intergovernmental Partnership	○	●	○
	• E.O. 12988, Civil Justice Reform	○	●	○
	• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	●	○
	• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify) E.O. 12898			

AGENCY:	Nuclear Regulatory Commission
TITLE OF ACTION:	Use of Alternative Source Terms at Operating Reactors
LEVEL OF SIGNIFICANCE:	Non-major rule
UPCOMING ACTION:	Final rule
AGENCY IDENTIFICATION:	RIN 3150-AG12
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DESCRIPTION Of ACTION:

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Submission of Federal Rules Under the Congressional Review Act

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3150 AG12

5. Major Rule ○ Non-major Rule ●

6. Final Rule ● Other ○ ______

7. With respect to this rule, did your agency solict public comments? Yes ● No ○ N/A ○

8. Priority of Regulation (fill in one)

● Economically Significant; or Significant; or Substantive, Nonsignificant

○ Routine and Frequent or Informational/Administrative/Other (Do not complete the other side of this form if filled in above.)

9. Effective Date (if applicable) 30 days following notice in Federal Register

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Submitted by: ______________________ (signature)

Name: Dennis K. Rathburn

Title: Director, Office of Congressional Affairs

For Congressional Use Only:

Date Received: ____________

Committee of Jurisdiction: ____________________

3/23/99

24722

	Yes	No	N/A
A. With respect to this rule, did your agency prepare an analysis of costs and benefits?	●	○	○
B. With respect to this rule, by the final rulemaking stage, did your agency			
1. certify that the rule would not have a significant economic impact on a substantial number of small entities under 5 U.S.C. § 605(b)?	●	○	○
2. prepare a final Regulatory Flexibility Analysis under 5 U.S.C. § 604(a)?	○	●	○
C. With respect to this rule, did your agency prepare a written statement under § 202 of the Unfunded Mandates Reform Act of 1995?	○	●	○
D. With respect to this rule, did your agency prepare an Environmental Assessment or an Environmental Impact Statement under the National Environmental Policy Act (NEPA)?	●	○	○
E. Does this rule contain a collection of information requiring OMB approval under the Paperwork Reduction Act of 1995?	●	○	○
F. Did you discuss any of the following in the preamble to the rule?	●	○	○
• E.O. 12612, Federalism	○	●	○
• E.O. 12630, Government Actions and Interference with Constitutionally Protected Property Rights	○	●	○
• E.O. 12866, Regulatory Planning and Review	○	●	○
• E.O. 12875, Enhancing the Intergovernmental Partnership	○	●	○
• E.O. 12988, Civil Justice Reform	○	●	○
• E.O. 13045, Protection of Children from Environmental Health Risks and Safety Risks	○	●	○
• Other statutes or executive orders discussed in the preamble concerning the rulemaking process (please specify)			

E.O. 12898

3/23/99

AGENCY:	Nuclear Regulatory Commission
TITLE OF ACTION:	Use of Alternative Source Terms at Operating Reactors
LEVEL OF SIGNIFICANCE:	Non-major rule
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ESTIMATED DATE OF ISSUANCE:	October 1999
STATUTORY OR JUDICAL DEADLINE:	N/A

DESCRIPTION Of ACTION:

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Attachment 7

Draft

Public Announcement

Jtg
G:\DPR\srcetrm.

July 27, 1999 (9:20AM)

OPA

D R A F T

(Source:Holahan memo & draft SECY paper of 7/7/99)

NEW NRC REGULATION TO PERMIT NUCLEAR POWER PLANTS TO CHANGE ACCIDENT ANALYSES OF PUBLIC RADIATION DOSE

The Nuclear Regulatory Commission has amended its regulations to permit nuclear power plant licensees to take advantage of updated research findings on estimated public radiation doses from reactor accidents.

The new rule will permit these licensees to use what is known as an alternative "source term" for the accident analysis on which plant design and operations are based, replacing a source term that has been in effect for the past 37 years. Experience from the 1979 Three Mile Island accident and research that followed it have made this change possible.

"Source term" is the technical name for the calculation of the speed, magnitude and chemical form in which the radioactive material produced by the atom-splitting process in a nuclear reactor would be released from the reactor to the containment if an accident occurred.

Nuclear power plants use the source term for analyzing possible accident consequences -- including potential radiation dose to the public from leakage out of the containment into the environment -- and factor that analysis into plant design and operation.

All currently operating nuclear power plants were licensed on the basis of a source term published in 1962 by the Atomic Energy Commission, NRC's predecessor agency. That procedure assumed an immediate release of radioactive materials to the containment during a severe accident, including a substantial amount of radioactive iodine which could cause thyroid cancer.

But what occurred in the Three Mile Island accident, in addition to extensive research which followed it, suggests that a release into the containment would be phased, rather than immediate. Revised source terms published by NRC in 1995 reflected those findings.

The rule now being adopted will permit utilities with nuclear power plant operating licenses to replace the 1962-era source term in their licenses with a revised one. NRC believes this change can reduce an unnecessary burden on many licensees without compromising public health and safety, reduce worker radiation exposure, and improve overall safety. This regulation, however, is not intended to provide licensees with relief from NRC's emergency planning requirements.

Specifically, it is expected that such a change could cut down on occupational radiation exposures in activities such as the frequency of installation of charcoal filters, maintenance of certain containment isolation valves, and repairs to systems to maintain leak-rate limits that are overly restrictive in the light of the recent research. Cutting back on this unnecessary work also could lead to cost savings. Improvements in overall safety are also likely due to, for example, reduction in the loading of emergency diesel generators.

Licensees who wish to continue with their present source term can do so.

Along with the adoption of the new rule, NRC also has published for public comment a draft regulatory guide and a new section of the NRC Standard Review Plan, both of which are intended to give licensees guidance as to acceptable methods of complying with the new rule. The new regulation will take effect 30 days after its publication in a forthcoming edition of the Federal Register.

That Federal Register notice also will have more information about the new draft regulatory guide and Standard Review Plan section. Comments on the latter two documents should be submitted 75 days after the Federal Register notice's publication.

After this rule was published in draft form in March, NRC received seven comment letters, all of which were supportive. The NRC staff also conducted two workshops, in April and in June, with interested stakeholders to discuss this rulemaking.

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