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

S.F. LaVie
NRR/DSSA/SPSB

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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. <u>Existing Regulatory Framework</u>

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 to applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under Part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under Part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67. 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. <u>Backfit Considerations</u>

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 (including those with renewal licenses for which the initial operating license was issued prior to January 10, 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. <u>Impacts</u>

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. <u>Backfit Considerations</u>

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 is applicable to the following classes of licensees:

    - applicants for and holders of construction permits and operating licenses under Part 50 who apply on or after January 10, 1997,

    - applicants for design certifications under Part 52 who apply on or after January 10, 1997,

    - applicants for and holders of combined licenses under Part 52 who do not reference a standard design certification, and

    - holders of operating licenses using an alternative source term under § 50.67.

  The final rule requires these licensees to show compliance with the 0.05 Sv (5 rem) TEDE dose criterion. There are currently no applicants for or holders of licenses in the above four categories.

5. <u>Impacts on NRC Resources, Other NRC Programs, and Other Agencies</u>

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 June 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. <u>Applicability to Other Accidents</u>

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.