U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[1]

Secondary - ~~Effluent Treatment Systems Branch (ETSB)~~Plant Systems Branch (SPLB)[2]

I. AREAS OF REVIEW

This SRP section covers the review of the radiological consequences of a postulated fuel handling accident. The purpose of the review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The review includes the following:

1. The review is concerned with the selection of values of plant parameters for use in calculating the radiological consequences of a fuel handling accident, and the selection of the dose computation model, including assumptions of transport mechanisms and rates from the fuel handling area to the atmosphere, breathing rates, dose conversion factors, and other data that may affect the calculated dose.

2. The calculated doses are compared with the appropriate exposure guidelines to determine the acceptability of the exclusion area boundary and low population zone (LPZ) boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

3. The containment ventilation system is reviewed with respect to its function as a dose mitigating engineered safety feature (ESF) system for a fuel handling accident inside the containment, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. ~~The closure times for the isolation valves in the lines are reviewed by the Containment Systems Branch (CSB).~~[3]

~~4. The Effluent Treatment Systems Branch (ETSB) reviews, under SRP Section 6.5.1, the ESF atmosphere cleanup systems used to mitigate the radiological consequences of accidents. ETSB provides the filter efficiencies for the ESF systems to AEB for use in the analysis of the radiological consequences. This is a secondary review effort by ETSB.~~[4]

~~5. Auxiliary Systems Branch (ASB) reviews under SRP Section 9.4.2 the design and operation of the spent fuel pool area ventilation system.~~[5] ~~The AEB reviewer verifies with the ASB the assumptions for the system with respect to its function as a dose mitigating system. This is a coordinating review function.~~[6]

~~6. The movement of heavy loads (i.e., loads heavier than the combined weight of a spent fuel assembly and the fuel handling tool) or of irradiated fuel in the spent fuel pool and over the open reactor vessel is reviewed by ASB under SRP Sections 9.4.1 and 9.4.2. An analysis of the radiological consequences may be required for such drops of heavy objects if more than one fuel assembly can be damaged. The need for such calculation is determined by ASB who will advise AEB (note: the radiological consequences of a fuel cask drop in which the fuel inside the cask is damaged is reviewed by the AEB under SRP Section 15.7.5).~~[7]

4. The PERB also reviews the sections listed below and utilizes the information obtained to support the review of SAR Section 15.7.4.

   a. SRP Section 9.4.2 for verification with the SPLB assumptions for the spent fuel pool area ventilation system with respect to its function as a dose mitigating system.

   b. SRP Section 15.7.5 for the radiological consequences of a fuel cask drop in which the fuel inside the cask is damaged.

   c. SRP Section 2.3.4 for determining the acceptability of the X/Q values for the short term atmospheric dispersion factors.

Review Interfaces[8]

The PERB coordinates other branch evaluations that interface with the overall review as follows:

1. The Plant Systems Branch (SPLB)

   a. Reviews the ESF atmosphere cleanup systems used to mitigate the radiological consequences of accidents and reviews and verifies the acceptability and efficiencies of the system under SRP Section 6.5.1.

   b. Provides the filter efficiencies for the ESF systems to PERB for use in the analysis of the radiological consequences.

   c. Reviews the design and operation of the spent fuel pool area ventilation system under SRP Section 9.4.2.

   d. Reviews the movement of heavy loads (i.e., loads heavier than the combined weight of a spent fuel assembly and the fuel handling tool) or of irradiated fuel in the spent fuel pool and over the open reactor vessel under SRP Sections 9.1.4 and 9.1.5.

   e. Determines the need for an analysis of the radiological consequences that may be required for drops of heavy objects if more than one fuel assembly can be damaged and advise PERB of the need. (Note: the radiological consequences of a fuel cask drop in which fuel inside the cask is damaged is reviewed by the PERB under SRP Section 15.7.5.)

2. The Containment Systems and Severe Accident Branch (SCSB) reviews the closure times for the isolation valves in the containment purge/vent lines for those plants that purge or vent the containment during fuel handling operations.

3. If the values proposed by the applicant for gap activity or peak assembly power are less than those in Regulatory Guide 1.25, the Reactor Systems Branch (SRXB) is requested to review these values in a coordinated review effort. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."[9]

4. Upon request, the Mechanical Engineering Branch (EMEB) verifies the number of rods assumed damaged for the fuel handling accident both within the spent fuel pool storage area and inside containment.[10]

For those areas of review identified as part of the primary responsibility of other branches, the acceptance criteria and methods of application are contained in the referenced SRP section.[11]

II. ACCEPTANCE CRITERIA

The ~~AEB~~PERB[12] acceptance criteria for this SRP section are based on requirements of ~~(Ref.1)~~[13] 10 CFR Part 100 with respect to the calculated radiological consequences of a fuel handling accident and ~~(Ref.2)~~[14] General Design Criterion 61 with respect to appropriate containment, confinement, and filtering systems. Specific criteria necessary to meet the requirements are:

1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 750 mSv (75 rem)[15] for the thyroid and 60 mSv (6 rem)[16] for the whole-body doses.

2. The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, ~~(Ref. 2)~~[17] "Fuel Storage and Handling and Radioactivity Control," with respect to appropriate containment, confinement, and filtering systems.

3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25,~~(Ref.3)~~[18] with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4. The source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."[19]

4. An ESF-grade atmosphere cleanup system is required for the spent fuel storage area to reduce the potential radiological consequences.

5. The containment design is acceptable with respect to a postulated fuel handling accident if it possesses the capability for prompt radiation detection by use of redundant radiation monitors and automatic isolation if fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system). An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations.

Technical Rationale[20]

The technical rationale for application of these acceptance criteria to reviewing the radiological consequences of fuel handling accidents is discussed in the following paragraphs:[21]

1. Compliance with 10 CFR Part 100, section 100.11, limits the total radiation dose to the whole body and to the thyroid at the exclusion area and low population zone boundaries given a fission product release from a postulated accident.

10 CFR Part 100 is applicable to SRP Section 15.7.4 in that the radiological consequences of a postulated fuel handling accident, appropriate containment, confinement, and filtering systems must be considered in the review of SRP Section 15.7.4 to determine the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries. Regulatory Guide 1.25 provides additional guidance in meeting these requirements. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."

Meeting this requirement provides assurance that radiation dose to the whole body and to the thyroid at the exclusion area and low population zone boundaries are well within the exposure guidelines contained in paragraph 100.11 of 10 CFR Part 100.[22]

3. Compliance with GDC 61 requires, in part, that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

   GDC 61 is applicable to SRP Section 15.7.4 in that the SRP covers the review of the radiological consequences of a postulated fuel handling accidents that could involve damage to spent fuel. Such postulated accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. Therefore, appropriate containment, confinement, and filtering systems are designed and reviewed to reduce potential fuel handling accident radiation doses to well within acceptable limits.

   Meeting the requirements of GDC 61 provides assurance that the radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building provide adequate safety during normal operations and during postulated accidents.[23]

III. <u>REVIEW PROCEDURES</u>

The reviewer selects and emphasizes specific aspects of this SRP section as are appropriate for the particular plant. The judgment on which areas need to be given attention and emphasis are determined by the similarity of the information presented by the applicant to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The relevant portion of Chapter 15 of the applicant's safety analysis report (SAR) are reviewed to determine the values of those fuel parameters which affect fission product release and fuel pool iodine decontamination factors, including the maximum fuel rod pressurization, peak linear power density for the highest power assembly, maximum centerline operating fuel temperature for the peak assembly, average burnup for the peak assembly, and minimum water depth between the top of any damaged fuel rods and the water surface.

2. The staff performs an independent dose calculation using the assumptions in Regulatory Guide 1.25. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."[24] If the values proposed by the applicant for gap activity or peak assembly power are less than those in Regulatory Guide 1.25, the ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[25] should be requested to review these values in a coordinated review effort. If other factors are used that are less conservative than those recommended in Regulatory Guide 1.25, which may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants,"[26] ~~are used,~~ Reference 4 should be consulted to determine if an adequate basis for the proposed deviation exists.

   Three important parameters affecting the radiological consequences of a fuel handling accident are not covered in Regulatory Guide 1.25. These are the reactor design (stretch) power level, the earliest time after reactor shutdown that fuel handling operations can commence, and the number of fuel rods assumed to be damaged in a fuel handling accident. The reactor design power level is obtained from Section 1.1 or Chapter 15 of the SAR.

   Unless the applicant proposes otherwise, the standard technical specification (STS) values for minimum time to fuel handling are used to determine the earliest time after shutdown for fuel handling. ~~(Current STS values are 24 hours for a boiling water reactor, 72 hours for a CE design and 100 hours for other pressurized water reactors).~~[27] The applicant should provide in the SAR conservative analyses of the number of rods assumed damaged both for the spent fuel storage area and inside containment, and the Mechanical Engineering Branch ~~(MEB)~~(EMEB)[28] should be requested to verify the number of rods assumed damaged. Reference 6 may also be consulted in this regard.

3. Fuel handling accident in fuel buildings: The applicant's SAR is examined to ~~assure~~ensure[29] that an ESF atmospheric cleanup system is included in the design of the fuel storage facility to mitigate the radiological consequences of a fuel handling accident. Verification of acceptability and efficiencies of the atmosphere cleanup system are provided by the ~~ETSB~~SPLB[30] through the review of SRP Section 6.5.1. The reviewer should examine those pertinent aspects of the accident, especially with regard to the operational modes of the ventilation systems and location and response time of the radiation detectors to ~~assure~~ensure[31] that any accidental release will be detected in sufficient time to be appropriately ducted and exhausted via ESF filters.

4. Fuel handling accident inside containment: The systems to mitigate the consequences are reviewed. If an applicant proposes that fuel handling will occur only when the containment is isolated, no radiological consequences need be calculated. If fuel handling operations occur only when the containment is exhausted to the environment via an ESF filter system, the radiological consequences should be calculated giving appropriate credit for this system. If the containment will be open during fuel handling operations, as with a containment purge exhaust system, the reviewer should verify that a

prompt radiation detection and automatic containment isolation capability are provided and that the resulting doses are within the acceptance criteria given in subsection II.1 above.

For a plant design with the containment open during the fuel movements, a review should be made of the applicant's analysis. This should include an examination of the type, location and redundancy of the radiation monitors intended to detect an activity release inside the containment and verification that detection is followed by automatic containment isolation. The reviewer should assess the time required to isolate the containment. This should include the instrument line sampling time (where appropriate), detector response time and containment purge isolation valve actuation and closure time. The containment is considered isolated only when the purge isolation valves are fully closed. The applicant's analysis should be reviewed regarding the travel time of any activity release starting from its release point above the refueling cavity or transfer canal and including travel time in ducts or ventilation systems up to the inner containment purge isolation valve.

The time required for the release to reach the inner isolation valve is compared ~~to~~with the[32] time required to isolate the containment. If the time required for the release to reach the isolation valve is longer than the time required to isolate containment, then essentially no release to the atmosphere occurs, and the reviewer's assessment should reflect this. If the time required for the release to reach the isolation valve is less than that required to isolate containment, and no mixing or dilution credit can be given, the reviewer should assume that the entire activity release escapes from the containment in evaluating the consequences. Claims for credit for dilution or mixing of a release due to natural or forced convection inside containment are reviewed and assessed. References 4 and 5 should be consulted and used by the reviewer for guidance in estimating dilution and mixing. Where mixing and dilution can be demonstrated within containment, the radiological consequences will be reduced by the degree of mixing and dilution occurring prior to containment isolation.

5. The atmospheric dispersion factors; X/Q values, to be used in analyzing the consequences of the accident are provided by the assigned meteorologist.

6. The doses calculated by the applicant and independently by the staff are compared ~~to~~with[33] acceptance criteria in subsection II. If the results of the dose calculations indicate the dose guideline values may be exceeded, alternatives which would reduce the doses to an acceptable level are examined and explored with the applicant (e.g., increased distance, better filters).

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[34]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided by the applicant and the staff independent dose calculations support conclusions of the following type, to be included in the staff's safety evaluation report (SER):

> The staff finds that the applicant has provided an adequate system to mitigate the radiological consequences of a postulated fuel handling accident inside the containment and in the fuel building. The staff concludes that the fuel handling system meets the relevant requirements of General Design Criterion 61. The staff further concludes that the distance to the exclusion area and to the low population zone boundaries for the (INSERT PLANT NAME) site, in conjunction with the operation of dose mitigating ESF and implementation of plant procedures, are sufficient to provide reasonable assurance that the calculated offsite radiological consequences of a postulated fuel handling accident are well within the 10 CFR Part 100 exposure guidelines.
>
> The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at the (INSERT PLANT NAME) facility meet the requirements of General Design Criterion 61 with respect to radioactivity control; (2) the staff review of the applicant's assumptions and analyses of the radiological consequences from the fuel handling accident; and (3) the staff's independent analyses using the assumptions in Regulatory Guide 1.25, Portions C.1.a through C.1.k. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."[35]

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff’s evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[36]

V. IMPLEMENTATION

The following provides guidance to applicants and licensees regarding the staff's plans for using this SRP Section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[37] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[38]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 100, Paragraph ~~11~~100.11,[39] "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

3. 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."

4. Evaluation of Fission Product Release and Transport for a Fuel Handling Accident by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, revised October 5, 1971.

5. Industrial Ventilation/A Manual of Recommended Practice - American Conference of Governmental Industrial Hygienists.

6. Long Island Lighting Co., et al., Docket No. STN 50-516/517, Further additional supplemental testimony on contention I.D.2 (Spent Fuel Handling Accident) by Walter L. Brooks, et al.

7. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."[40]

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Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB name and abbreviation	Changed PRB to Emergency Preparedness and Radiation Protection Branch (PERB).
2.	Current SRB name and abbreviation	Changed SRB to Plant Systems Branch (SPLB).
3.	SRP-UDP format item	Relocated under "Review Interfaces" and reworded as item 2.
4.	SRP-UDP format item	Relocated under "Review Interfaces" and reworded as items 1.a and 1.b.
5.	SRP-UDP format item	Relocated under "Review Interfaces" and reworded as item 1.c.
6.	SRP-UDP format item	Relocated under "Review Interfaces" and reworded as item 1.d.
7.	SRP-UDP format item	Relocated first sentence to "Review Interfaces" and reworded as item 1.d. Relocated the remainder to "Review Interfaces" and reworded as item 1.e.
8.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW and organized as numbered paragraphs to describe how PERB reviews aspects of the radiological consequences of fuel handling accidents under other SRP sections and how other branches support the review of the radiological consequences of fuel handling accidents. Review interfaces were excerpted from subsection I, AREAS OF REVIEW, and subsection III, REVIEW PROCEDURES.
9.	SRP-UDP format item	Excerpted from REVIEW PROCEDURES, subsection III.2, and added "as modified by revised source terms for the gap activity from Table 3.12 of draft NUREG-1465," which reflects Integrated Impact No. 842.
10.	SRP-UDP format item	Excerpted from REVIEW PROCEDURES, subsection 2, paragraph 3.
11.	SRP-UDP format item	Revised to reflect the current format when the SRP section contains review interfaces.
12.	Current PRB name and abbreviation	Changed PRB to PERB.
13.	SRP-UDP format item	Deleted (Ref. 1).
14.	SRP-UDP format item	Deleted (Ref. 2).
15.	SRP-UDP format item, convert to metric units	Converted rem to mSv.
16.	SRP-UDP format item, convert to metric units	Converted rem to mSv.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
17.	SRP-UDP format item	Deleted (Ref. 2).
18.	SRP-UDP format item	Deleted (Ref .3).
19.	Integrated Impact No. 842	Added: The source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff-approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." (This addition accommodates Integrated Impact No. 842.)
20.	SRP-UDP format item, develop "Technical Rationale"	Added "Technical Rationale" to ACCEPTANCE CRITERIA subsection and organized in numbered paragraph form to describe the basis for referring to the GDC.
21.	SRP-UDP format item, develop "Technical Rationale"	Added lead-in sentence for "Technical Rationale."
22.	SRP-UDP format item, develop "Technical Rationale"	Added technical rationale for 10 CFR 100.11.
23.	SRP-UDP format item, develop "Technical Rationale"	Added technical rationale for GDC 61.
24.	Integrated Impact No. 842	Added: The applicants source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." (This addition accommodates Integrated Impact No. 842.)
25.	Current review interface branch abbreviation	Changed review interface branch to SXRB.
26.	Integrated Impact No. 842	Added: Which may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." (This addition accommodates Integrated Impact No. 842.)
27.	Editorial	Deleted a phrase containing information from a primary document that is subject to change in this secondary document.
28.	Current SRB abbreviation	Changed SRB to EMEB.
29.	Editorial	Changed "assure" to "ensure."
30.	Current review interface branch abbreviation	Changed review interface branch to SPLB.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
31.	Editorial	Changed "assure" to "ensure."
32.	Editorial	Changed "compared to" to "compared with" to accommodate scientific usage.
33.	Editorial	Changed "compared to" to "compared with" to accommodate scientific usage.
34.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
35.	Integrated Impact No. 842	Added: The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing, may be modified by other Staff approved source terms and methodologies such as those contained in NUREG-1465. (This addition accommodates Integrated Impact No. 842.)
36.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
37.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
38.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
39.	Editorial	Corrected paragraph numbering.
40.	Integrated Impact 842	Added Reference 7, NUREG-1465.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
704	Consider future work to revise RG 1.25 to incorporate the results of the side-by-side comparison.	None. This impact was not processed further pending action tracked by IPD-7.0 Form 15.7.4-1.
842	Revise acceptance criteria and review procedures to incorporate the application of revised source term data.	SRP 15.7.4, Subsections II.3, III.2, IV, and VI.7
1232	Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.	This is a placeholder integrated impact.