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Draft Reactor Accident Analysis Modernization (RAAM) Report

July 2024

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REACTOR ACCIDENT ANALYSIS MODERNIZATION

1.0 INTRODUCTION

The objective of the Reactor Accident Analysis Modernization (RAAM) project is to investigate current reactor accident analysis and review methods and to propose ways, if possible, to modernize the approach while ensuring appropriate conservatisms in the regulatory accident analysis and review. A key goal is to assess if reactor accident analyses can be simplified, streamlined, or otherwise improved.

This work was guided by a charter endorsed by the NRR Divisions of Safety Systems (DSS), Risk Assessment (DRA), Non-Power Production and Utilization Facilities (DANU), and New and Renewed Licensing (DNRL) as well as the Division of Risk Analysis (DRA) in RES (ML23172A305). This report was developed by a working group consisting of five Senior Level advisors (SLSs), a licensing Branch Chief, and a Senior Reliability & Risk Analyst. Subject matter experts with experience in performing various reactor analyses and risk analyses were consulted by the core team members with defined assignments on specific review areas of shorter duration. As defined in the RAAM charter, the working group was responsible for developing a set of high level recommendations on how risk-informed decision-making can be better used to enhance the licensing review of design basis accident safety analyses while maintaining a clear licensing basis for regulated facilities.

When incorporating risk-insights to develop recommendations, the RAAM working group used several key PRA policies that provided guidance to the staff. These policies include the 1985 Severe Reactor Accident policy statement (50 FR 32138; August 8, 1985) which described the policy related to accidents more severe than the design basis accidents, and the 1995 PRA policy (60 FR 42622; August 16, 1995) which provides the overall policy on the use of PRA methods in nuclear regulatory activities. The Severe Accident Policy Statement recognizes the usefulness of PRAs in identifying severe accident vulnerabilities and providing additional insights to ensure that nuclear power plants do not result in an undue risk to public health and safety. The PRA policy statement acknowledges that PRA technology should be used in all regulatory matters to the extent supported by the state-of-the-art and that its use can reduce unnecessary conservatisms.

The team leveraged information gathered from current rules and regulations along with recent agency licensing and guidance development activities. This effort was focused on the transient and accident analyses as found in NUREG-0800 Chapter 15 under Part 50 and Part 52 licensing applications. This scope included underlying rules, regulations, and guidance which supports licensing reviews of accident analyses.

The main activities assigned to the RAAM Working Group include the following:

- Developing a mapping for the rules, regulations, and guidance associated with transient and accident analyses,
- Identifying the current ways in which risk information is used in accident analyses,
- Considering of lessons learned from previous applications of risk-informed approaches in accident analysis (e.g., use of best-estimate methods, credit for containment overpressurization, etc.),
- Identifying best practices from the international regulatory community and other domestic regulators which use risk information in their safety evaluations,
- Identifying rules, regulations, and guidance which could be modified to support increased use of risk-informed approaches,
- Identifying potential options to increase the use of risk information in accident analyses and evaluations, consistent with Commission expectations stated in the PRA policy statement and supported by additional Commission direction and policy decisions,
- Identifying existing risk information initiatives that could support this effort to modernize accident analyses,
- Identifying areas where review efficiency can be increased by using risk-insights to determine the need for additional confirmatory analyses or in-depth reviews/audits,
- Identifying potential impacts from increasing the use of risk informed approaches on rules and regulations. This assessment may include identification of new or revised analysis metrics and acceptance criteria, new analysis approaches (considering multiple rather than single failures), more explicit treatment of uncertainties, etc.

The RAAM working group developed a mapping of SRP Chapter 15 analyses to their respective regulatory requirements and identified a set of potential areas that might be suitable for modernization using risk-informed principles. The potential areas for modernization consideration are described in Section 2.

2.0 POSSIBLE ACTIONS FOR CONSIDERATION

The RAAM Working Group identified several potential modernization activities that are described in this section. These items reflect the Working Group’s understanding of current licensing issues and opportunities as of early CY2024. These are binned to several application focus areas, including analysis acceptance criteria, LOCA assumptions, single failure, credit for non-safety-related SSCs, and potential Licensing Modernization Project (LMP) applications:

Focus Area	Item	Report Section	Preliminary Working Group Recommendation
Anticipated Operational Occurrence Acceptance Limits	Redefine Acceptable Fuel Design Limits (SARDLs/SAFDLs)	2.1	Additional Study Needed (e.g., Public Meeting). Potentially useful long term

Non-Safety-Related SSCs	Risk-Informed Guidance for Crediting Non-Safety-Related SSCs	2.2	Pursue – High Priority
Loss of Coolant Accidents	Use of an Alternate Criteria to 95/95 for LOCA	2.3	Pursue – High Priority
	Redefine Large Break LOCA to Beyond Design Basis Event	2.4	Pursue – Under Increased Enrichment Rulemaking
	Reconsideration of LOCA Break Locations	2.5	Defer – potentially limited interest from industry
Single Failure	Risk-inform Single Failure Criteria	2.6	Pursue
	Define Single Passive Failures for Fluid Systems	2.7	Pursue
Environmental Qualification	Risk-Inform EQ Radiological Requirements	2.8	Defer
Design Basis Accidents	Increase the Coherence and Consistency of DBA Radiological Consequence Analysis	2.9	Continue to pursue resolution of DPO 2020-002 and 2021-001 with high priority
Licensing Modernization Project Applications	Use of LMP Results to Focus Staff Reviews	2.10	Pursue
	Use of Event Sequence Frequencies to Risk-Informed Accident Analysis	2.11	Pursue

2.1. REDEFINE ACCEPTABLE FUEL DESIGN LIMITS IN GENERAL DESIGN CRITERION 10

2.1.1. Description

GDC 10 currently requires all LWRs to define Specified Acceptable Fuel Design Limits (SAFDLs) which are not to be exceeded during any condition of normal operation, including AOOs. NUREG-0800 Section 4.2 provides additional guidance by defining acceptable LWR

SAFDLs which are based on the concept of retaining cladding integrity for normal operation and AOOs. Effectively, this approach means that the SAFDLs are set such that no radiological release is predicted for normal operation and all AOOs.

Another approach would be to instead require LWRs to define a Specified Acceptable Radiological Release Dose Limit (SARRDL) and provide justification for this limit based on the specific reactor and fuel design. This allows for a more risk-informed approach by considering potential reactor or fuel-specific design aspects which might either reduce the likelihood of a radiological release event (e.g. increased defense-in-depth) or else mitigate the consequences of a release. It is important to note that applicants wishing to use this approach would need to provide the radiological release dose limit and provide the supporting analysis demonstrating their ability to calculate fuel failures and radiological consequences for their given fuel and plant design.

This approach would lean on recent advanced non-LWR experience, including guidance found in RG 1.232, "Guidance for Developing Principal Design Requirements for Non-Light-Water-Reactors" (ML17325A611). Actual implementation of this approach would center around a revision to NUREG-0800 Section 4.2 to build in an option of a SARRDL approach in addition to the current SAFDL approach.

This approach would leverage recent work done for non-LWR fuel designs and would provide some flexibility to the current NUREG-0800 Chapter 4.2 limits which could allow for increased innovation. For example, a new ATF design might be developed which would drastically reduce the release of radiological particles which might in turn mean that the failure of a limited number of fuel rods would be acceptable from a safety perspective. It might also be shown that for a future LWR SMR design, the source term within a single fuel element might be small enough that it would not cause a safety concern. Similarly, new reactors might have additional defense-in-depth measures which are not considered in the current NUREG-0800 Section 4.2 guidance which was based on existing LWRs of the time.

2.1.2. Associated Regulatory Requirements

The associated regulatory requirements include:

> Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

2.1.3. Associated Guidance Documents

- NUREG-0800 Section 4.2, "Fuel System Design", provides the guidance to reviewing fuel system designs and ensuring compliance with GDC 10

- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” provides guidance to developing principal design criteria for non-LWRs, including approaches to ARDC 10 (the advanced reactor equivalent to GDC 10).

2.1.4. Potential Implementation Options

The RAAM working group believes that this change would be a significant change from past precedence; therefore the RAAM working group believes that further discussion with OGC would be required to determine if this change requires a rule change to GDC 10. Regardless of the rulemaking implications, this option would require a change in NUREG-0800 Section 4.2 to include an option similar to the SARRDL concept from RG 1.232. Updates to the NUREG-0800 typically go out for public comment and ACRS review, so the timeframes for rulemaking and NUREG-0800 updates are not significantly different. Therefore, either implementation option would be considered a relatively high level of effort. However, rulemaking would involve Commission approval steps which may require additional levels of effort and time. The RAAM WG believes that the SARRDL concept would be limited to LWR fuel designs that offer safety advantages compared to current LWR designs, such as fuel systems using TRISO or possibly a metallic fuel concept.

Another option for consideration would be to investigate the criteria listed in NUREG-0800 Section 4.2 to determine the amount of margin to likely fuel failure and provide recommendations as to whether the margin can safely be reduced. This option would seek to develop more performance-based criteria in SRP 4.2 that would preserve the intention of the SAFDLs, but provide additional flexibility to licensees and applicants. This would likely not require any rulemaking but would probably also have minimal impact for the industry as well.

2.1.5. Potential Impact for Operating LWRs

It is expected that the change from SAFDLs to SARRDLs as described above will not have a large impact on the operating fleet with standard fuel designs. This is due to the fact that with no other changes to reactor or fuel design, this change in regulatory approach would simply reduce defense-in-depth and would be difficult to justify based on the current role of fuel cladding in the safety analysis of the existing fleet. If an operating reactor were to utilize a future ATF design, then it is possible that the fuel design itself would provide an additional defensive layer which could then be used to justify the SARRDL approach for an existing reactor (e.g. a metallic fuel concept, or a TRISO compact based fuel inside a Zr-based cladding, etc.).

It is also worth noting that GDC 10 and this modification only applies to normal operation and AOOs. Therefore, for this change to have a transformative impact on a plant, it would need to currently be limited by an AOO as described by their FSAR. However, as plants move towards longer cycle lengths and increased enrichment, the flexibility provided by the SARRDL approach might have additional benefits to a licensee not currently reflected in their FSAR.

2.1.6. Potential Impact for New LWRs

New LWR designs leave open the possibility of increasing defense-in-depth either through novel fuel designs (e.g. metallic fuel in a Zr-based cladding) or a reactor design which can theoretically include additional radiological migration barriers not currently existing in the operating fleet. Also, some SMR's have comparatively short fuel designs which might result in fuel pins with significantly lower radiological source terms than would be expected in a currently operating LWRs. Therefore, it is possible that new LWR designs might be a good fit for the SARRDL concept. It is hard to gauge the level of impact a SARRDL approach would have on a new reactor design application, but it is possible to envision scenarios in which a new design could justify multiple fuel failures, which could greatly impact their approach to Chapter 15 analyses.

2.1.7. RAAM WG Assessment

Based on its preliminary assessment, the RAAM working group views this effort to have minimal impact on existing LWR reactors unless a new ATF design potentially leads to significant changes to the role of fuel in the safety analysis, at which point this effort could potentially have a medium impact. There are potential benefits for new LWRs (e.g. Holtec, NuScale, etc.) but it will be very important for clear guidance to be provided to ensure that applicants do not incorporate the SARRDL concept in an unsafe manner. Since the change will be a relatively significant one from the standpoint reactor fuel licensing, OGC guidance will be necessary. It is anticipated that rulemaking related to GDC 10 would likely be required given the strong precedence the use of SAFDLs has played in licensing. Further, it should be recognized that the fuel cladding is safety related barrier and is heavily relied upon in the safety analyses; therefore, any change to use a SARRDL criteria will need to ensure an adequate level of safety is maintained.

Regardless of the use of SARRDLs, the RAAM WG believes developing more performance-based options for SAFDLs in SRP 4.2 is recommended.

2.1.8. Recommendation

The RAAM working group recommends deferring implementation of this option until a later time. While it is feasible to do now, there isn't currently a big interest from the industry and it would likely require a substantial effort to implement. If some of the long-term ATF concepts become more likely, this recommendation can be reconsidered.

2.2. DEVELOP RISK-INFORMED GUIDANCE FOR CREDITING NON-SAFETY RELATED SSCS IN ACCIDENT ANALYSES

Under the current definition of safety related in 50.2, any SSCs credited with providing mitigation functions during a design basis event must be designated as "safety-related" and are subject to the quality assurance requirements in 10 CFR 50, Appendix B. This option would examine the possibility of crediting SSCs that are not "safety-related" in design basis event analysis provided certain conditions were met.

2.2.1. Description

NRC regulations (e.g., 10 CFR 50.2) require that SSCs be classified as “safety-related” if they are relied upon to remain functional during and following design basis events to assure:

- the integrity of the reactor coolant pressure boundary
- the capability to shut down the reactor and maintain it in a safe shutdown condition; or
- the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in the regulations (e.g., limit dose at the exclusion area and low population zone to less than 25 rem TEDE).

In accordance with 10 CFR 50.54(a)(1), each nuclear power plant subject to the quality assurance criteria in appendix B of this part shall implement the quality assurance program described or referenced in the safety analysis report. As noted in the introduction to 10 CFR 50, Appendix B, the requirements of that appendix apply to all activities affecting the safety-related functions of those structures, systems, and components.

In addition to “safety-related” SSCs, other regulatory requirements specific special treatment requirements, including quality assurance, design, and monitoring processes, for certain equipment that is not safety-related. These regulations include the following:

- 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 1, “Quality standards and records,” requires structures, systems, and components “important to safety” (which encompasses a broader range of equipment than “safety-related”) shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- 10 CFR 50.44, “Combustible gas control for nuclear power reactors,” Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur. RG 1.7, “Control of Combustible Gas Concentrations in Containment,” provides quality assurance requirements for non-safety-related SSCs required to meet the requirements of 10 CFR 50.44.

- 10 CFR 50.48, “Fire protection,” requires that each holder of an operating license or combined license have a fire protection plan that satisfies GDC 3, “Fire Protection.” GDC 3 requires, in part, that fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety Guidance for the implementation of a fire protection QA program is provided in RG 1.189, “Fire Protection for Nuclear Power Plants.”

- 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” Which requires certain ATWS equipment to perform in a “reliable manner” and included specific reference to issuance of QA guidance for non-safety related equipment. This guidance, contained in Appendix A to RG 1.155, “Station Blackout,” specified an acceptable QA program for nonsafety-related equipment used to comply with 10 CFR 50.63.

- 10 CFR 50.63, “Loss of all alternating current power,” requires that LWR nuclear power polants be able to withstand for a specified duration and recover from a station blackout of a specified duration. Although quality assurance requirements are not explicitly contained in the rule language, RG 1.155, ”Station Blackout,” include guidance for quality assurance requirements for nonsafety-related SSCs that are relied upon to meet the rule.

- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” requires that certain actions be taken if performance monitoring indicates that maintenance are not sufficient to maintain the performance and condition of specified safety-related and non safety-related SSCs, that the impact of maintenance on availability and reliability be appropriately balanced, and that the risk associated with maintenance activities be appropriately managed.

- 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” allows a graded approach to the quality assurance requirements for safety-related and nonsafety-related SSCs. Of particular note, the requirements for safety-related, but low safety significant, SSCs (i.e., RISC-2) can be relaxed compared to high safety significant SSCs (i.e., RISC-1). Additionally, nonsafety SSCs with high safety significance (i.e., RISC-3) are treated in a manner commensurate with safety-related high safety significant SSCs.

It is well recognized that the terms “safety-related” and “important to safety” appear throughout the regulations. The usage of these terms has, at times, caused confusion about the appropriate regulatory requirements associated with each category. It is important to recognize that the term “important to safety” includes both safety-related SSCs and nonsafety-related SSCs that perform safety functions. The use of these terms was discussed in Generic Letter 84-01, "NRC use of the terms, "Important to Safety" and "Safety Related"," which noted that NRC regulatory jurisdiction involving a safety matter is not controlled by the use of terms such as "safety-related" and "important to safety". GL 84-01 further clarified that:

> *"…nuclear power plant permittees or licensees are responsible for developing and implementing quality assurance programs for plant design and construction or for plant operation which meet the more general requirements of General Design Criterion for plant equipment "important to safety," and the more prescriptive requirements of Appendix B to 10 CFR part 50 for "safety-related" plant equipment."*

In addition, during the 1990s, the NRC identified the need to enhance the treatment of certain SSCs used in advanced light water reactor (ALWR) designs, particularly designs that had increased reliance on passive safety features. In SRM SECY 95-0132, the Commission approved staff proposals related to the Regulatory Treatment of Non-Safety Systems (RTNSS) and the Design Reliability Assurance Program (D-RAP). These programs include the following elements:

- RTNSS – The staff noted that ALWR designs relied on safety-related passive features as the primary means to perform design basis safety functions such as core and containment cooling. In addition, these designs also included active, non-safety-related systems that provided defense-in-depth capabilities and served as an initial line of defense to reduce challenges to the passive safety features. It was noted that residual uncertainties associated with passive systems (due to factors like low driving forces from gravity driven circulation) increased the importance of active, non-safety-related, features. The staff concluded, and the Commission approved, that these front-line active systems do not need to meet all safety-related criteria, but the staff will expect "a high level of confidence that active systems which have a significant safety role are available when challenged." In collaboration with EPRI, the staff developed criteria applicable to these active systems that will ensure that they meet their reliability and availability goals. The non-safety systems subject to RTNSS include SSCs relied upon to meet beyond design basis requirements such as ATWS and SBO; the Commission safety and containment goals; and prevent significant adverse system interactions.

- D-RAP – The RAP applies to those plant SSCs that are risk-significant (or significant contributors to plant safety) as determined by using a combination of probabilistic, deterministic, or other methods of analysis used to identify and quantify risk such as the design certification PRA. RAP is intended to provide assurance that the reactor is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for these risk-significant SSCs. The Commission approved the staff recommendation that RAP program elements be included within 10 CFR 52 design specific certification rulemaking. Key elements of the RAP include programmatic controls to ensure that design and construction are consistent with key risk insights and other QA activities related to design and construction.

Staff review guidance for the RTNSS program can be found in SRP Section 19.3, "Regulatory Treatment of Non-Safety Systems (RTNSS) for Passive Advanced Light Water Reactors," and for the RAP program in SRP Section 17.4, "Reliability Assurance Program (RAP)."

In summary, the agency has imposed additional programmatic requirements on nonsafety-related equipment as needed to provide assurance that these SSCs can reliably perform their safety function. The scope of equipment includes design basis equipment needed to support ATWS, combustible gas control, maintenance rule, GDCs, and other SSCs shown to be risk significant. Therefore, while there is strong precedent for acknowledging that key safety functions can be performed by nonsafety-related equipment, the staff has imposed graded programmatic requirements when needed to demonstrate their reliability and availability. A similar extension might be made for SSCs credited in design basis events under certain circumstances. For example, the staff has approved use of the nonsafety-related condenser as a deposition and holdup volume for fission products under certain conditions for design basis consequence analysis[1].

2.2.2. Associated Regulatory Requirements

Regulations associated with this issue are described in the preceding sections.

2.2.3. Associated Guidance Documents

Guidance documents associated with this issue are provided in the preceding section.

2.2.4. Potential Implementation Options

In order to provide credit for nonsafety-related equipment in design basis event analysis, several considerations must be resolved:

- Clear identification of the types of functions that can be performed by nonsafety-related equipment.
- How the performance goals (e.g., reliability, availability) would be identified for nonsafety-related equipment and how performance against these goals would be maintained.
- The programmatic requirements that would apply to nonsafety-related equipment credited in reactor accident analysis to ensure reliability and availability goals are met.

It may be necessary to issue license or applicant specific exemptions pursuant to 10 CFR 50.12 to implement credit for nonsafety-related equipment that would otherwise be categorized as

[1] See staff Safety Evaluation for GE Topical Report NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems"

safety related under 10 CFR 50.2. Alternately, rulemaking could be performed to clarify these requirements and provide direct reference to alternate treatment requirements such as RTNSS and RAP.

2.2.5. Potential Implication for Operating LWRs

Since the operating reactors are unlikely to request modifications to their already approved licensing and design bases to credit nonsafety-related SSCs in reactor accident analyses, the expected impact is minimal.

2.2.6. Potential Impact for New LWRs

Based on activities relating to new as well as advanced reactors (including advanced non-light water reactors), creating a formal regulatory structure to credit non-safety related systems in accident analysis can improve the efficiency of the staff review while ensuring an appropriately level of safety is maintained.

2.2.7. RAAM WG Assessment

The expected benefit of clarifying the role of nonsafety-related SSCs is expected to be medium for new reactor license applicants and low for operating reactors. It is expected that there would be high levels of interest in this item by new reactor designers. The resource costs to develop new policy and guidance is expected to be medium to high and require extensive stakeholder interactions and socialization within the staff. Case-by-case reviews can continue to be done, utilizing other regulatory mechanisms such as the 10 CFR 50.12 exemption process as needed to address review-specific issues.

2.2.8. Recommendation

Although it is expected that there will be medium to high resource expenditures associated with this item (largely to support significant internal and external stakeholder), the working group believes that the potential high benefits for new reactor licensing, as well as enhancing the regulatory stability for both operating and new reactors, warrants pursuit of this option high priority.

2.3. USE OF ALTERNATE CRITERIA TO 95/95 FOR LOSS OF COOLANT ANALYSES

2.3.1. Description

The wording within 10 CFR 50.46 stipulates "…when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded." Historically, the interpretation of "high level of probability" has been to use a 95% probability at a 95% confidence level when developing empirical evaluation models for LOCAs. This has been supported through decades of precedence. The RAAM working group notes however that 10 CFR 50.46 does not define "high level of probability" and therefore there is room for interpretation.

Based on internal discussions, the working group feels that another way to approach this interpretation of the rule is to use risk insights to justify a lower probability and/or confidence level if appropriate. For example, one hypothetical approach would be to set a probability (or consequence) criterion by which the event itself (e.g. an instantaneous pipe rupture leading to a loss of coolant) would be measured. If the event probability were to be demonstrated to be below this criterion, then a lower acceptance criterion (e.g. 1σ instead of 2σ) could be used to develop the empirical formulas in an applicant's evaluation model.

It should be noted that during a recent ACRS SC meeting on FRAMATOME's ARITA topical report, the ACRS chair brought this topic up and indicated that some sort of relaxation on the traditional 95/95 approach could potentially be appropriate.

2.3.2. Associated Regulatory Requirements

This modification to the review approach would most directly be associated with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors".

2.3.3. Associated Guidance Documents

There are various guidance documents which could be impacted by this change. The working group has currently identified the following:

1. NUREG-0800 Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary"
2. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance"

2.3.4. Potential Implementation Options

The working group concludes that it is possible to classify this proposed change as modification to guidance. This would therefore not require rulemaking. However, it is recognized that LIC-200 provides process guidance for modifications to NUREG-0800 which are not overly dissimilar to the rulemaking process. Therefore, a change to guidance provided in NUREG-0800 section 15.6.5 would still require a significant effort. Regulatory Guides (e.g. RG 1.157) have a similarly long change process. Additionally, the RAAM working group would recommend consultation with OGC to determine if this change would require additional regulatory process control beyond that typically associated with NUREG-0800 modifications.

The RAAM working group notes that with the current Increased Enrichment Rulemaking effort well underway, there is also a window of opportunity to include updates to specific guidance depending on which alternative is chosen as part of the rulemaking effort. Therefore, the working group recommends to closely coordinate with the Increased Enrichment (IE) Rulemaking (3150-AK79, SECY 21-0109) team to include the recommended modifications within guidance associated with the rulemaking package, assuming that the chosen IE Rulemaking alternative aligns with this approach. It is also recognized that NUREG-0800 and RG 1.157 are guidance which identifies an acceptable means to demonstrate compliance with regulations, which leaves open the possibility for the staff to consider applications with similar approaches during licensing reviews, while the guidance is being revised.

2.3.5. Potential Impact for Operating LWRs

Lowering the 95/95 data-fit related to LOCA modelling will have some impact on the operating fleet of LWRs but is not expected to have a dramatic impact. That being said, there is an industry-wide push to increase cycle length to 24 months. For LOCA-limited licensees, this change could potentially allow the licensee to increase cycle length to 24 months whereas the traditional approach would not.

2.3.6. Potential Impact for New LWRs

It is expected that the impact on new LWRs will be similar as to that for the operating fleet. A noted significant difference is that new reactors are in the process of designing their ECCS and might reduce the ECCS capacity for handling LOCAs based on the revised guidance. This would not necessarily be a regulatory issue since they would still have to show that they meet safety requirements, but it would potentially have the effect of decreasing the design of ECCS performance capabilities as compared with a design that otherwise would be built using a 95/95 acceptance criterion.

2.3.7. RAAM WG Assessment

The RAAM WG feels that this option would have a medium impact overall, but it would be highly plant-dependent. Some LOCA-limited plants might be able to increase cycle length proceed with a power uprate that they otherwise would not be able to support. The effort involved for this option might be comparatively lower since it likely would not require rulemaking, although further consultation with OGC would be recommended.

2.3.8. Recommendation

The RAAM working group recommends pursuing this effort unless IE Rulemaking effort supersedes its need. While it wouldn't likely make a large difference for most licensees, it could have a large operational impact on a few. Additionally, there appears to be a likelihood of acceptance from ACRS for this type of change and the overall effort might be less than some other options.

2.4. REDEFINE A POSTULATED LARGE BREAK LOSS-OF-COOLANT ACCIDENT FROM A DESIGN BASIS EVENT TO A BEYOND DESIGN BASIS EVENT

2.4.1. Description

For postulated loss-of-coolant accidents, for which the initiating event frequency falls below a threshold value, licensees and applicants would be able to treat the event as a beyond design basis event (BDBE). Such events could be analyzed using methods that are more consistent with Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," of the Standard Review Plan (SRP) or a final safety analysis report (FSAR), rather than using guidance provided in NRC Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," or in SRP Sections 6.3, "Emergency Core Cooling System," 15.0.2, "Review of Transient and Accident Analysis Method," and 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."

This is essentially closely related to Alternative 2 of Appendix F to the Increased Enrichment Rulemaking Regulatory Basis. If desired, the RAAM WG could investigate a small combination of IE Rulemaking Alternative 2 with a bounding approach to Alternative 3; however, this could be a difficult task.

2.4.2. Associated Regulatory Requirements

The impacted regulatory areas are as follows:

> 10 CFR 50.2, "Definitions," insofar as it establishes that safety-related structures, systems and components (SSCs) are those SSCs that are relied upon to remain functional during and following design basis events.
>
> 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," paragraph (a)(1)(i), insofar as it requires that ECCS cooling performance must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated, and

that when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of § 50.46, there is a high level of probability that the criteria would not be exceeded.

10 CFR 50.46, paragraph (c)(1), insofar as it establishes that loss-of-coolant accidents are hypothetical accidents that would result from the loss of reactor coolant from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the larges pipe in the reactor coolant system.

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 35, "Emergency Core Cooling," insofar as it requires the provision of a system to provide abundant emergency core cooling, the safety function of which is to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The definition of "safety-related SSCs" would likely not need to change; however, this relaxation, interpretation, or re-interpretation of existing requirements could allow licensees to demonstrate successful mitigation of those LOCAs that are re-categorized as BDBEs using non-safety-related SSCs.

The requirements set forth in 10 CFR 50.46(a)(1)(i), requiring assurance that the most severe postulated loss-of-coolant accidents are calculated, and establishing a high level of probability that the criteria contained in 10 CFR 50.46(b) are not exceeded, may not be met if a licensee or applicant wishes to exclude larger break loss-of-coolant accidents from a design basis.

2.4.3. Associated Guidance documents

Guidance provided in NRC Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," describes approaches acceptable to the NRC staff in evaluating ECCS performance in compliance with the requirements in 10 CFR 50.46(a)(1)(i). Regulatory Position 3.1 states, "The calculations performed should be representative of the spectrum of possible break sizes from the full double-ended break of the largest pipe to a size small enough that it can be shown that smaller breaks are of less consequence than those already considered." Regulatory Position 3.4.1 reinforces this concept.

An enabling edit to this guidance could introduce an upper limit to the maximum analyzed break size that is based on an initiating event frequency. Specific guidance concerning acceptable ways to estimate such a frequency could also be included.

Section 6.3, "Emergency Core Cooling System," of the SRP provides guidance concerning the ECCS design and hardware but defers to SRP Chapter 15 for the ECCS performance evaluation. It is anticipated that implementing this modernization option will require minimal, if any, revision to SRP Section 6.3.

15.0.2, "Review of Transient and Accident Analysis Method," and 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."

2.4.4. Potential Implementation Options

Likely, this option could be implemented with minor edits to 10 CFR 50.46(a)(1)(i) and to 50.46(c) to revise the definition of the postulated loss-of-coolant accident and to change the assurance statement. The definition could remove "up to and including a double-ended rupture…" and replace with a frequency-based break size. The assurance statement ("sufficient to provide assurance that…" could similarly be revised to include language about a most severe LOCA excluding the least likely events. This might look like the changes to 50.46 and appendix k that were made to enable measurement uncertainty recapture uprates.

Alternatively, a more holistic approach could look like those changes proposed in accord with the 10 CFR 50.46a rulemaking from the early 2000s[2].

Regardless of path, the RAAM WG feels that if requested, the RAAM WG would be supporting the IE Rulemaking group as opposed to creating a new independent effort.

2.4.5. Potential Impact for Operating LWRs

For plants that are design limited by ECCS (i.e., LOCA is the most limiting design basis event), this modification would open operating flexibilities to gain additional fuel economy. Approximately one dozen operating PWRs might fall in this category. Several BWRs, especially BWR/3 and early-generation BWR/4s, also fall in this category.

Additionally, for plants wishing to apply additional capital investment (e.g., major plant modifications), this change could make improvements like power uprates more feasible or economically appealing.

For all plants, this change could allow a treatment to certain ECCS that is different from current practice and requires less ongoing surveillance and maintenance.

2.4.6. Potential Impact for New LWRs

[2] As discussed in 81 FR 69446 (https://www.govinfo.gov/content/pkg/FR-2016-10-06/pdf/2016-24189.pdf), the staff discontinued the 10 CFR 5046a rulemaking as part of the AIM 2020 shed process (see SECY 16-0009 and its associated SRM). The staff concluded, that based on interactions with the nuclear industry, there were concerns with the potential implementation burden of the rule.

This approach would allow applicants to propose designs with less robust and less costly ECCS in comparison the LOCA mitigation capability of the currently operating LWRs. Since LWR designers generally consider PRA insights during the design process, these designers may be able to demonstrate that a less robust and less costly ECCS does not result in plant designs that are less safe than the plants that are presently operating in US. It could enable different categorization of LOCA mitigating systems. Additionally, this chance could allow existing, proposed designs to increase system capabilities (e.g., equivalent to operating plant power uprate).

2.4.7. RAAM WG Assessment

The RAAM WG believes that there is industry interest in this topic based on the level of participation in the Increased Enrichment (IE) Rulemaking process. The impact on licensees could be high depending on various factors like desire for uprates and whether or not the plant is LOCA-limited.

The impact on new LWR applicants could be substantial as they could design within the revised regulations which could change their approach to licensing. The RAAM WG does not recommend creating a separate effort and instead for the RAAM WG to be a resource to assist the IE Rulemaking WG in addressing this issue. The resource requirements for this item are expected to be high.

2.4.8. Recommendation

The RAAM working group and steering committee recommends pursuing this issue under the ongoing IE Rulemaking effort. Depending on the eventual outcome of the IE Rulemaking, the need to pursue further enhancements in this area for either operating or new reactors could be reevaluated. This effort would likely have appreciable benefits for the industry, especially those licensees wishing to pursue power uprates or go to a 24-month refueling cycle.

2.5. RECONSIDERATION OF LOSS OF COOLANT ACCIDENT BREAK LOCATIONS

2.5.1. Description

10 CFR 50.46 requires LWR reactors to be designed with an Emergency Core Cooling System (ECCS) capable of maintaining fuel within specified criteria during a Loss of Coolant Accident (LOCA). One part of this rule states: "…must be calculated for a number of postulated loss-of-

coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated". The intent of the rule is to ensure that the ECCS is sufficiently sized to reliably bring the reactor to a safe state after a loss of coolant accident up to and including the size of a double-ended guillotine break of the largest piping penetration.

Some applicants have voiced frustration during LOCA methodology reviews and feel that the NRC staff spends significant review time asking questions about hypothetical break locations that do not have an appreciable impact on safety. While the rule does dictate that the most severe postulated LOCAs are considered, the RAAM working group considers this an area where the use of risk insights can help to better define how to implement this requirement. An impact of considering different reactor coolant system break locations is a potential change to the postulated break size and sizing of the ECCS.

2.5.2. Associated Regulatory Requirements

10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," which stipulates that LOCA analyses must bound the possible break sizes, locations, etc.

2.5.3. Associated Guidance Documents

NUREG-0800 Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary" which provides guidance related to LOCA analyses which are used to demonstrate compliance with 10 CFR 50.46.

2.5.4. Potential Implementation Options

The RAAM working group investigated two possible solution options to this problem. The first is to instruct the technical reviewers to only review the technical sufficiency of the analyzed postulated breaks as identified by the applicant. The second option is to increase the use of risk information in the technical reviews.

The RAAM working group concludes that relying solely on the applicant to determine the limiting break location would not align with 10 CFR 50.46, which directly calls out break locations as being part of the review to provide reasonable assurance that the most severe postulated LOCA is analyzed. To enact this type of change would require rulemaking and would also potentially allow an applicant to either inadvertently or intentionally choose a non-limiting break location which would result in an unanalyzed condition which could result in an ECCS design incapable of adequately providing necessary core cooling. Therefore, the RAAM working group does not recommend this approach.

The second option is to better risk inform the reviews such that sufficiency of break locations would be considered alongside the likely impact of a different break location and available margin as presented (and reviewed) in the analysis. Existing guidance (i.e. NUREG-0800

Section 15.6.5) could be modified to more clearly instruct the reviewer to consider the specifics of the unanalyzed break location and available margin before requiring the applicant to perform additional analyses. This could also lead to consideration of non-piping break locations if they were risk significant for the design. The RAAM working group recommends this approach.

2.5.5. Potential Impact for Operating LWRs

If a licensee or applicant submits a high-quality application, the main impacts of either of these options would be related to review time. In the first option, the reviewer would no longer confirm that the break location was the limiting one, which would reduce the overall review time. In the second option, the reviewer would "right-size" the review which could potentially reduce overall review time.

2.5.6. Potential Impact for New LWRs

It is expected that the impacts of this change on new LWRs would be the same as that for operating LWRs.

2.5.7. RAAM WG Assessment

The RAAM WG believes that additional information would be needed to better understand the extent of this problem. If this industry criticism appears to be widespread, then the RAAM WG would recommend a modification to NUREG-0800 15.6.5 to better provide guidance regarding the application of risk insights within the review. Based on the current understanding of the issue, the RAAM WG believes this concern to be somewhat limited and the impacts of the changes to be relatively low. Resource requirements are estimated to be medium to high.

2.5.8. Recommendation

The RAAM working group recommends deferring this option until a later time. While there are some examples of where increased use of risk information for break location determination might be beneficial, there isn't currently a big interest from the industry as a whole and the RAAM Working Group does not see this as an area that would result in significant efficiency gains. If we note an increase in feedback from applicants related to this topic, this option can be revisited.

2.6. RISK INFORM APPLICATION OF THE SINGLE ACTIVE FAILURE CRITERION

Allow relaxation in the application of the single failure criterion when other reliability considerations demonstrate high confidence in accomplishment of system safety functions.

2.6.1. Description

The single failure criteria for active SSCs has been traditionally applied in a deterministic manner generally without regard to the underlying accident sequence frequency or SSC reliability. This helps to ensure that safety systems include a sufficient level of redundancy to ensure that safety functions can be accomplished. However, by accounting for the specific accident sequence context, the probability of failure of an SSC that would normally be subject to single failure assumptions could be combined with other sequence specific reliability attributes to allow relaxation of the application of single failure in certain circumstances. This item would support a more risk-informed treatment of single failure which could allow exceptions to the single failure criteria while still ensuring high confidence in safety system capabilities.

2.6.2. Associated Regulatory Requirements

There are multiple references to the single failure criteria throughout 10 CFR Part 50, which are intended to ensure the sufficient reliability, redundancy, independence, and testability of safety functions. Although the term "single failure" is not defined in 10 CFR 50.2, "Definitions," a definition is provided in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." Specifically, Appendix A, defines "single failure" as[3]:

> *"…an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."*

The specific references to "single failure" in 10 CFR 50 include the following:

- 10 CFR 50.34(f)(3)(vi) requires plant designs with external hydrogen recombiners, redundant dedicated containment penetrations are provided so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.

- 10 CFR 50.49(e)(3) requires that SSCs subject to environmental qualification (which includes safety-related electrical equipment) that the assumed chemical effects include the

[3] This definition includes a footnote stating that single failures of passive components in electric systems should be assumed in designing against a single failure but that the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure "are under development". The issue related to single failure of a passive component in a fluid system is also highlighted in the Introduction to 10 CFR 50, Appendix A, where it is noted that several specific design requirements for SSCs important to safety have not been suitably defined, including consideration of the need to design against single failures of passive components in fluid systems important to safety. This issue is addressed further in Section 3.8 of this report.

most severe chemical spray environment that results from a single failure in the spray system.

- Several specific GDCs reference the application of the single failure criteria, including:
  - GDC 17 (Electric power systems) requires the onsite electric power supplies to have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
  - GDC 21 (Protection system reliability and testability) requires protection systems to be designed with redundancy and independence sufficient to assure that no single failure results in loss of the protection function.
  - GDC 24 (Separation of protection and control systems) requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.
  - GDC 25 (Protection system requirements for reactivity control malfunctions) requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.
  - GDC 34 (Residual heat removal), GDC 35 (Emergency core cooling), GDC 38 (Containment heat removal), GDC 41 (Containment atmosphere cleanup), and GDC 44 (Cooling water) require suitable redundancy in components and features shall be provided to assure that the system safety functions can be accomplished, assuming a single failure.
- 10 CFR 50, Appendix K, “ECCS Evaluation Models,” requires consideration of the most damaging single failure of ECCS equipment.

- 10 CFR 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” requires separation between two water supply systems so that failure of one supply will not result in failure of the other supply.

Collectively, these requirements ensure that safety-related and important to safety SSCs are designed with adequate redundancy and reliability. Additionally, single failure requirements also address the ability to perform surveillance or other testing on SSCs while maintaining the ability to accomplish the safety functions.

2.6.3. <u>Associated Guidance Documents</u>

There is limited formal guidance for applying the single failure, but RG1.53, “Application of the Single-Failure Criterion to Safety Systems,” and SRP Chapter 15.0, “Introduction - Transient and Accident Analyses,” provide some guidance. RG 1.53, which is applicable to electrical systems (e.g., power, instrumentation, and control safety systems) and augments the codes and standards requirements specified in 10 CFR 50.55a(h), “Protection and safety systems,” by endorsing the more recent version of IEEE Standard 279-2000. SRP chapter 15.0, Section I.4 states that the “performance of each credited protection or safety system is required to include the effects of the most limiting single active failure.”[4] SRP Chapter 15.0, Section 1.6.B, further states that the reviewer verifies that only safety-related systems or components are specified for use in mitigating AOO and postulated accident conditions, and the effects of single active failures in those systems and components are included. Other sections of the SRP direct the reviewer to verify that the single failure criteria is met when performing system level reviews. For example, SRP Chapters 6.2.2, “Containment Heat Removal Systems,” 6.2.4, “Containment Isolation System,” 6.3-6, “Emergency core Cooling System,” 8.3.1, “A-C Power Systems (Onsite),” 8.3.2, “D-C Power Systems (Onsite),” and 9.2.1, “Station Service Water,” all include guidance related to application of the single failure criteria for system level reviews.

More specific guidance on application of the single failure criteria can be found in several Commission level documents. For example, SECY 77-439 (ML060260236) provides an overview of how the staff applied the single failure criteria in the reactor safety review process. SECY 77-439 noted the application of the single failure criteria involves a systematic search for potential single failure points as a means to search for design weaknesses. SECY 77-439 then described how the staff applied the single failure criteria for electrical and mechanical systems. SECY 77-439 noted that the single failure criteria was “developed without the benefit of numerical assessments on the probabilities of component or system failure” but that it is not assumed that any conceivable failure could occur. The paper then provides examples of where single failure had not applied, including for reactor vessels or certain types of structural elements within systems, that when combined with other unlikely events, are not assumed to fail “because the probabilities of the resulting scenarios of events are deemed to be sufficiently small that they need not be considered.” With regard to electrical failures, the staff stated that there is not a distinction between failures of active and passive electrical components and all such failures must be considered in applying the single failure criteria. SECY 77-439 further stated that the practice for passive failures in fluid systems was limited assuming failure of a pump or valve seal during long-term cooling, and that more significant failures such as pipe breaks were not considered due to the compound probability of the sequence being “sufficiently small” that they can be discounted. Check valves were a somewhat unique case in that SECY 77-439 classified the failure of “a simple check valve to move to its correct position when required” as an example of a passive failure which was not required to be considered as a single failure. It was also noted that consideration of the likelihood of a single failure could change the classification of a design basis event (e.g., consideration of multiple failures may change an AOO to a postulated accident) and result in less stringent acceptance criteria. SECY 77-439 also notes that certain factors, such as common mode failure and system functional dependencies or interactions, can undermine the redundancy provided by application of the single failure criteria. The paper concluded by noting that application of the single failure criteria

[4] SRP Chapter 15.0, Section VI, “Definitions,” defines “single failure” as “[a]n occurrence that results in a component’s loss of capability to perform its intended safety functions.”

should continue “pending any long-term wide-scale incorporation of reliability and risk assessment methodology into the licensing process.”

In SECY 94-084 (ML003708068), the staff recommended that the Commission approve an exception to the SECY 77-439 single failure guidance related to check valves in passive advanced LWR safety systems. Specifically, the staff recommended that the Commission approve a proposal to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration. This change was motivated by the recognition that certain passive systems may operate with reduced differential pressures, which may reduce the reliability of check valves in these systems. The staff further noted that in “determining an exemption to single failure consideration for a particular check valve application, the plant designer shall perform a comprehensive evaluation of check valve test data or operational-data for the similar check valve designs in similar applications and operating environments to demonstrate that the reliability of the particular check valve application is such that the probability of failure is comparable to those of passive components.” In SRM SECY 94-084, the Commission approved the staff’s recommendation.

Several of the risk-informed concepts introduced in SECY 77-439 were further developed in SECY 05-0138 (ML051950619). Specifically, SECY 05-0138 requested that the Commission approve release of a draft report (ML051950625) on risk-informed options for the single failure criteria to gather additional stakeholder feedback. The draft report described the following options: (1) continuation with status quo; (2) elimination of sufficiently unlikely sequences from design basis event analysis and adding sequences with multiple failures when sequence frequency is high; (3) apply the single failure criteria based on the safety significance of systems, including consideration of non-safety systems; and (4) allow for risk-informed treatment of the levels of redundancy and diversity for safety functions and establish targets for unreliability which would include consideration of initiating event frequency. In SRM SECY 05-0138, the Commission approved public issuance of the draft single failure report and directed the staff to proceed with rulemaking activities (e.g., issuance of an advanced notice of proposed rulemaking (ANPR)) that consider the spectrum of issues relating to risk-informing reactor requirements.. In SECY 07-0101, the staff summarized stakeholders comments obtained from the ANPR process and recommended that further rulemaking activities to risk inform 10 CFR 50 be deferred until additional insights were obtained from expected licensing activities for the Next Generation Nuclear Plant (NGNP) and/or the Pebble Bed Modular Reactor (PBMR). The Commission approved deferral of further rulemaking activities in SRM SECY 07-0101 and directed the staff to publish the technology neutral framework for licensing (later published as NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing” in 2007). This work has not been restarted, but many elements of the draft proposed 10 CFR Part 53 (SECY 23-0021) built off the risk-informed concepts considered in this earlier rulemaking activity. Of note, the Licensing Modernization Process (LMP) approach (NEI 18-04), replaces the single failure criteria with a probabilistic reliability criterion. The staff has endorsed use of NEI 18-04, with clarifications, in RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors." Further, the staff based portions of the draft proposed 10 CFR Part 53 rule on the LMP methodology as described and approved by the Commission in SECY 19-0117 and its associated SRM.

In SECY 19-0036 (ML19060A162), which addressed a specific application of the single failure criteria to an active component in the NuScale design, the staff recommended the implementation of a risk-informed option to the single failure criteria that was built upon the concepts described in SECY 77-439, SECY 94-084, and SECY 05-0138. Specifically, the staff recommended in Option 1 a proposed an approach that would "allow an exception for application of the SFC for an active component where it can be shown that, consistent with SECY-77-439, the inherent reliability of the component, combined with the frequency of challenge to the component results in scenario likelihood 'deemed sufficiently small that they need not be considered.'" In SRM SECY 19-0036, the Commission disapproved this staff recommendation and instead directed that staff to not consider an active failure of the subject component in the Chapter 15 analysis.

2.6.4. Potential Implementation Options

The implementation history of the single failure criteria makes it clear that the staff has generally considered the probability of failure of a component as an important consideration when applying the single failure criteria. Though referenced in the regulations, the staff has generally implemented the single failure criteria through policy decisions and guidance. Potential options to further enhance the single failure criteria include:

- Development of a risk-informed process to consider the inherent reliability of an SSC in combination with the frequency of challenge that would allow relaxation of the criteria if scenario likelihood was sufficiently low.

- Develop a more holistic treatment of single failure that would consider use of probabilistic reliability criterion and include consideration of multiple failures, including potential of common cause failures.

- Maintain status quo and consider application of the single failure criteria on a case-by-case basis.

It is expected that the non-status quo options would require Commission interactions, consistent with past practice, to implement.

2.6.5. Potential Impact for Operating LWRs

Given the maturity of operating reactor designs, it is expected that future changes to the single failure criteria would have limited benefit to the operating fleet. A potential exception to this may be in the area of design basis accident consequence analysis where the use of accident tolerant fuel (ATF) or increased fuel burnup may challenge control room and offsite dose limits. This could also be addressed under the issues addressed under Section 3.10.

2.6.6. Potential Impact for New LWRs

For non-status quo options, it is anticipated that there would be greater interest in the licensing of new LWRs since the final design can take advantage of increased flexibility associated with alternatives for the single failure. High reliability passive systems with limited active components may provide greater confidence in meeting designated safety functions than an active system with multiple support systems. Therefore, alternatives to the single failure may be warranted when system reliability can be demonstrated by alternate means.

2.6.7. RAAM WG Assessment

Development of risk-informed guidance for application of the active single failure criteria is expected to have low benefit for operating reactors and high benefit for new reactors. The resource cost is expected to be medium to support broad internal and external stakeholder engagment. The staff will be able to leverage previous work done on this issue, but full application will require stakeholder interactions and formalization of guidance documents, including stakeholder comment.

2.6.8. Recommendation

Although it is expected that there will be some significant resource expenditures associated with this item (largely to support significant internal and external stakeholder engagement) the working group believes that the potential benefits for new reactor licensing warrants pursuit of this option. It is expected that previous work in this area can be leveraged to mitigate resource impacts.

2.7. DEFINE SINGLE FAILURE CRTIERIA FOR PASSIVE SSCS IN FLUID SYSTEMS

Provide enhanced guidance for how the failure of passive SSCs in fluid systems should be addressed under the single failure criteria.

2.7.1. Description

The single failure criterion as defined in 10 CFR 50, Appendix A, applies to both active and passive SSCs. Although the criterion is applied for both passive and active electrical components, as noted in 10 CFR 50, Appendix A, “the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.” Within the context of the General Design Criteria, a fluid system is considered to be designed against an assumed single passive failure if the failure does not result in a loss of the capability of the system to perform its safety functions. Although 10 CFR 50, Appendix A, has contained the footnote that the conditions for single passive failures in fluid systems were “under development” since the regulation was first promulgated in February of 1971 (36 FR 3256, Feb. 20, 1971), the NRC has not provided additional requirements for passive failures in the regulations in the intervening 53 years. However, the preamble for the 1971 rule stated that the omission of this safety consideration “does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements.”

Compared to currently deployed active safety systems in operating LWRs, safety systems in new and advanced reactor designs will place increased reliance on the use of passive safety features. For example, the 2008 Commission policy on the regulation of advanced reactors (73 FR 60612) notes that the Commission expects “that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.” While system reliability in currently operating LWRs is dominated by active component failures, passive failures are expected to dominate newer systems that rely on natural buoyancy forces and other inherent mechanisms. Further, developers are including innovative new component types in their designs, such as “fluidic devices” that rely on passive features to support key safety features. For example, in the Kairos Hermes design, a fluidic diode is used to direct natural circulation flow when forced circulation is lost[5]. The APR1400 design similarly used a fluidic device to achieve a desired reactor coolant injection profile without the need for any active components[6]. Given the lack of operating experience with these types of devices, the failure modes and component reliability for these novel devices may not be fully understood. Further, the performance of passive safety systems may be more strongly impacted by minor design errors or changes in system fluid dynamic properties than more conventional active safety systems.

2.7.2. Associated Regulatory Requirements

The regulatory requirements for consideration of both active and passive failures are summarized in Section 3.5 of this report. In addition, 10 CFR 50.34(f)(2)(xxvi) requires certain operating reactor licensees to "provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident.” This requirement is related to post-TMI action item III.D.1 which is intended to design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident. While this requirement is not intended to support safety system operation, it does address potential radiological

[5] Development of the fluidic diode device is noted as an ongoing research and development activity by Kairos (see the staff’s safety evaluation ML23158A268).
[6] As noted in Chapter 6, “Engineered Safety Features,” of the staff’s APR1400 safety evaluation (ML18212A092), the fluidic device is installed in the bottom of the SIT to provide two operational stages of safety water injection into the RCS, large flow mode and small flow mode.

consequences associated with failure of systems that could carry radioactive materials outside of containment. This requirement can be addressed through the implementation of a leakage control or monitoring program.

Although not directly related to the application of the single failure criteria, 10 CFR 54, “Requirements for the Renewal of Operating Licenses for Nuclear Power Plants,” includes specific requirements related to the management of aging for structures and components that (1) perform their intended function without a moving parts or without a change in configuration or properties, and (2) are not subject to replacement based on a qualified life or specified time period. Of particular note, 10 CFR 54.21(i) provides examples of components and structures that do not rely on moving parts or changes in configuration or properties including pressure boundaries, ducts, piping, structures, cable trays, and cabinets. Although these examples are pertinent only for the purposes of identifying structures and components subject to aging management, these items are often referred to as passive equipment.

2.7.3. <u>Associated Guidance Documents</u>

In addition to the guidance for application of the single failure criterion noted in Section 3.5, additional guidance is provided in Chapter 6.3, “Emergency Core Cooling System,” of the Standard Review Plan (ML070550068):

> *The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident.*
>
> *A passive failure in a fluid system is a breach in the fluid pressure boundary or mechanical failure that adversely affects a flowpath. SECY-94-084 states the approved position that passive advanced light-water reactor designs need not assume passive component failures in addition to the initiating failure in the application of single-failure criterion to assure safety of the plant. In addition, the staff considers, on a long-term basis, passive component failures in fluid as potential accident initiators, in addition to initiating events. Check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) are considered components subject to single-failure consideration.*

SECY 77-439 noted that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed to assure safety of a nuclear power plant. However, the SECY provides further information on the treatment of single passive failures for emergency core cooling systems:

> *…it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the longterm cooling mode following a LOCA (24 hours or greater after the*

> *event) but not pipe breaks. No other passive failures are required to be assumed because it is judged that compounding of probabilities associated with other types of passive failures, following the pipe break associated with a LOCA, results in probabilities sufficiently small that they can be reasonably discounted without substantially affecting overall systems reliability.*

The current guidance on passive failures is based on currently operating LWR experience, which may not apply to new reactor designs. Further, the emerging importance of passive safety systems in new designs may increase the importance of passive failures, as risk may no longer be dominated by active system failures. In addition, continuing interest in long term operation and life extension beyond 60 years, may increase the risk importance of age related degradation for piping and other passive components to the overall risk profile of currently operating plants.

2.7.4. Potential Implementation Options

The current guidance for application of the single failure criteria for passive systems is limited and does not address the full scope of failures described in the 10 CFR 50, Appendix A, definition for single failure. Additionally, the increasing importance of passive systems in new reactor designs, where designs may not include active components, increases the importance of the reliability of passive features. For the operating fleet, the reduced importance of passive failures compared to active single failures, combined with the license renewal aging management programs provides confidence that passive equipment will continue to perform their intended function.

Potential implementation options include:

Evaluation of the need to address single passive failures in currently operating and new and advanced reactor designs. If this evaluation indicates the need for enhanced guidance, then the following options could be implemented:

- Rulemaking activities to update the language in 10 CFR 50, Appendix A, related to the treatment of passive failures
- Issuance of enhanced guidance (both Standard Review Plan and Regulatory Guidance) for both currently operating and new and advanced designs to address passive equipment failures.

The implementation of any of the above actions would not be intended to change the licensing basis for currently operating LWRs with regard to passive single failures.

2.7.5. Potential Impact for Operating LWRs

It is expected that this issue would have minimal impact on operating LWRs since the licensing basis for consideration of single passive failures has been established and plants have demonstrated that they do not have significant vulnerabilities in this area as a result of the IPE/ IPEEE program and other initiatives. However, significant changes to the current licensing basis for operating LWRs may require reconsideration of how passive failures are considered in the licensing basis.

2.7.6. Potential Impact for New LWRs

Given the increased reliance on passive safety features and new types of passive components, enhanced guidance for the treatment of passive failures would lead to a more stable and predictable licensing process. Further, the existing guidance for treatment of passive failures is intended for designs similar to the current operating fleet and may not adequately address new reactor designs.

2.7.7. RAAM WG Assessment

The WG believes this issue would address an important safety consideration that has been left unfinished since the initial issuance of the GDCs in 1971. This issue would provide a low benefit for operating reactors and potentially a high benefit for new reactors. However, the resource cost is expected to be medium to high since full implementation would require rulemaking to clarify the NRC position on passive single failure in the General Design Criteria. This may require additional policy clarifications and extensive stakeholder interaction to develop appropriate criteria that is applicable for new reactor designs. However, once completed, this activity would provide regulatory stability and predictability in the licensing process while ensuring an appropriate level of safety is met.

2.7.8. Recommendation

Although it is expected that there will be medium to high resource expenditures associated with this item (largely to support significant internal and external stakeholder engagement and potential rulemaking activities), the working group believes that the potential benefits for new reactor licensing and enhancements to regulatory stability warrants pursuit of this option.

2.8. RISK INFORM APPLICATION OF ENVIRONMENTAL QUALIFICATION RADIOLOGICAL REQUIREMENTS UNDER 10 CFR 50.49

Provide enhanced risk-informed guidance to reduce unnecessary conservatisms in the implementation of environmental qualification requirements of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," for radiation hazards based on use of more realistic radiological source terms.

2.8.1. Description

The requirements of 10 CFR 50.49(e)(4) require, in part, that the environmental qualification program include and be based on “the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional.” Although 10 CFR 50 does not specifically define “design basis accidents”, 10 CFR 50.49(b)(ii) includes “design basis accidents” within the defined scope of “design basis events” used to identify safety-related equipment. Other regulations in 10 CFR 50 (for example 10 CFR 50.67, “Accident source term”) also refer to the term “design basis accident”, but the usage of this term appears in differing contexts. For example, the use of the term in 10 CFR 50.67 includes a footnote noting that such “accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products”; however, a similar footnote does not appear in 10 CFR 50.49. As discussed in SECY 19-0079, “Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application,” 10 CFR 50.49 does not include specific language pertaining to the use of a core damage source term to assess radiation effects. However, power reactor license applicants have generally considered a core melt accident source term when assessing radiation environmental hazards. The use of core damage source term is consistent with guidance provided in NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” and RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.” However, use of a core melt source term for the purposes of environmental qualification may be overly conservative if this radiation hazard exceeds the radiation environment under which the equipment would be expected to remain functional. For example, under the definition of safety-related in 10 CFR 50.2, some safety-related equipment that is designed to mitigate design basis events may not need to be qualified for a core damage source term if the spectrum of design basis events mitigated by the equipment does not include core damage. Consistent with the approach outlined in SECY 19-0079 for environmental qualification for the NuScale design certification, this item would develop risk-informed guidance that better aligns radiological environmental qualification requirements to the expected hazard that SSCs included within the scope for 10 CFR 50.49 would experience during the design basis events they are used to mitigate.

2.8.2. Associated Regulatory Requirements

10 CFR 50.49 requires the establishment of a program for qualifying the electrical equipment defined within the scope of the rule (which is termed “electric equipment important to safety”). 10 CFR 50.49(b) defines the scope of “electric equipment important to safety” as safety-related electric equipment, nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety-related functions, and certain post-accident monitoring equipment specified in RG 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Revision 2. With respect to the radiation environment, 10 CFR 50.49(e)(4) requires that:

The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

It is noteworthy that this regulatory language permits the radiation environment to be graded commensurate with the environment that the "equipment is required to remain functional" rather than a specific severe accident source term. This is in contract with other areas of the regulations where the source term is intended include fission product release from the core and should be assumed to be associated with accidents that "result in substantial meltdown of the core" (e.g., 10 CFR 50.34(a)(1)(ii)(D) and associated footnote, 10 CFR 50.67(b)).

The requirements of 10 CFR 50.49 do not address the environmental qualification of equipment located in a mild environment, which is defined as an "environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences."

2.8.3. Associated Guidance Documents

RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 2, was issued in April 2023 to endorse, with clarifications, standard IEC/IEEE 6078-323, Edition 1, 2016-2. In addition, Revision 2 to RG 1.89 referred to the guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," as guidance for "accident radiological source terms." The reference to RG 1.183 updates the guidance contained in Revision 1, issued in 1984, which did not address use of an alternate source term (e.g., NUREG-1465).

RG 1.183 provides core fission product release fractions up to and including the early in-vessel release phase for LOCA and non-LOCA accident conditions. It should be noted that, consistent with the requirements of 10 CFR 50.46, a LOCA is not expected to result in substantial fission product release form the core. The source term specified in RG 1.183 is intended to support the evaluation of the "safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur" (10 CFR 50.34(a)(1)(ii)(D)). While it is recognized that electrical equipment needed to mitigate a substantial fission product release postulated pursuant to 10 CFR 50.34(a)(1)(ii) or 10 CFR 50.67 needs to be qualified for a radiation environment consistent with the RG 1.183 source term, there are safety related features that are not required to maintain functionality in this extreme environment. Therefore, use of source term representing "substantial meltdown of the core" may be overly conservative for some equipment.

The requirements for identifying equipment subject to environmental qualification contained in 10 CFR 50.49(b) include “certain post-accident monitoring equipment” along with a clarifying footnote that refers to RG 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Revision 2. While the footnote does not provide a regulatory requirement, the current version of RG 1.97 is Revision 5, issued in 2019. Of note, Revision 5, endorses IEEE 497-2016, which provides a less prescriptive, more performance-based approach. In addition, Revision 5 to RG 1.97 acknowledges a new category of monitored variable, Type F, which provides primary information to indicate fuel damage and the effects of fuel damage.

2.8.4. Potential Implementation Options

The rule language in 10 CFR 50.49 provides flexibility to define the radiation environment for qualification in a manner consistent with the “most severe design basis accident during or following which the equipment is required to remain functional”. However, current guidance for the radiation source term does not make a clear distinction between equipment that must remain functional following a substantial core melt and other equipment where a reduced radiation hazard could be assumed (e.g., coolant and/or gap release). Similar to the approach used for NuScale (as described in SECY 19-0079, a more graded, performance-based approach could be described in guidance documents to reduce unnecessary conservatisms where appropriate. Such guidance could also recognize the importance of post-accident instrumentation and their role in supporting FLEX and SAMG actions. Guidance updates could also ensure coherence in the environmental hazard levels for non-radiation challenges

2.8.5. Potential Impact for Operating LWRs

This item is expected to have limited applicability to the operating fleet since equipment is already qualified consistent with the current licensing basis. However, it is possible that replacement instrumentation or other upgrades could benefit from guidance changes if a reduced radiation hazard could be justified. Additionally, there has been some recent indication that deployment of accident tolerant fuel concepts may decrease the longevity of certain SSCs and require additional risk-informed flexibilities.

2.8.6. Potential Impact for New LWRs

This item may have a more significant impact for new reactor designs during the design and procurement stages as guidance changes could support the efficient review of alternate, but justifiable, radiation environments for certain electrical equipment. Of particular note, certain small modular reactors that include smaller volume containments (such as the NuScale design) may benefit since the concentration of the core melt within a smaller containment volume can increase the accident radiation exposure for new designs.

2.8.7. RAAM WG Assessment

This item is expected to provide a relatively low benefit, since this issue has not been identified as a significant obstacle to operating plants or new design applicants (with the exception of NuScale). For a small demand load, case-by-case reviews can be conducted. Full implementation would require revision to regulatory guidance documents along with associated stakeholder interactions and expected to be a medium resource cost.

2.8.8. Recommendation

Given the expected limited benefits to operating and new reactor licensing, combined with the potential resource needs, the working group recommends deferring this option. However, if industry interest increases in this option as a result of broad accident tolerant fuel deployment, this option can be revisited.

2.9. DEVELOP ENHANCED GUIDANCE TO INCREASE THE COHERENCE AND CONSISTENCY OF DESIGN BASIS ACCIDENT RADIOLOGICAL CONSEQUENCE ANALYSES

Design basis accident radiological consequence analyses supports the evaluation of engineered safety features and barriers used to mitigate release of fission products to the environment. Because significant fission product release from the core is an initial assumption of these assessment, the analysis relies on a unique combination of stylized bounding assumptions, licensing basis information, and risk insights. Balancing these various perspectives has, at times, led to confusion and incoherence in how these analyses are performed. Use of risk-informed insights can help provide a clearer technical basis for these assessments and ensure coherence and consistency in these evaluations.

2.9.1. Description

Design basis accident analysis is performed to assess postulated fission product release from a nuclear power plant, using the expected containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. The objective of these assessments is the demonstrate that offsite radiological releases do not result in significant doses to the public. Specific criteria contained in 10 CFR 50.34(a)(1)(ii)(D), and echoed in 10 CFR 50.67, require that:

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

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

Pursuant to 10 CFR 50.34 and 10 CFR 50.67, these assessments consider a fission product release from the core that is: …

> *….based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products [footnote 6 to 10 CFR 50.34(a)(1)(ii)(D)].*

The design basis accident analysis includes both stylized assumptions (e.g., a source term intended to bound a spectrum of potential credible accidents) and licensing basis assumptions related to the performance and capabilities of plant SSCs. Consistent with the 10 CFR 50.2 safety related definition and technical specification requirements in 10 CFR 50.36, the design basis radiological consequence analysis generally considers only mitigation by safety related equipment and other constraints imposed by technical specifications.

In addition, GDC 19, "Control room," requires "[a]dequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident." An evaluation of control room dose is also included in the requirements of 10 CFR 50.67(b)(2)(iii). The control room dose limits do not specifically require mitigation to rely solely on safety-related equipment, but other considerations (e.g., need for operators to perform mitigation actions from the control room) may result in the need for safety related dose mitigation equipment. Additionally, while GDC 19 does not include a footnote referencing the use of a core melt source term, 10 CFR 50.67(b)(2) does include reference to a footnote that the evaluation for offsite and control room radiological consequences be based on "substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

Balancing the use of stylized bounding assumptions, licensing basis information, and risk insights has, at times, led to significant differences in opinion on how to best demonstrate the regulatory safety case. As a result, two Differing Professional Opinions have been evaluated since 2020 regarding what assumptions should be made when performing these analyses. These DPO's addressed different aspects of the design basis accident radiological consequence evaluation:

- DPO-2021-001 (ML23240A717) - This DPO raised concerns about the issuance of a license amendment approving use of the alternate source term at the James A. FitzPatrick nuclear power plant. Key issues raised in the DPO include use of fission product removal models known to be incorrect, consideration of alternate radionuclide leakage paths, and crediting nonsafety-related equipment. After careful consideration of the issues raised in the DPO, the EDO directed NRR to: (1) take appropriate regulatory action to ensure FitzPatrick's compliance with 10 CFR 50.67 and resolve the licensing clarity issues for the AST license amendment; and (2) develop an implementation plan for the other recommendations included in the DPO Appeal Panel Analysis Report.

- DPO-2020-002 (ML21060A972): The DPO submittal raised concerns with certain inconsistencies in the assessment of the radiological consequences of design basis accidents. Specific concerns focused the use of stylized assumptions intended to explain the presence of a severe accident source term rather than deterministically imposing the source term[7] and crediting the operation of safety-related SSCs consistent with standard practices. An Ad Hoc Panel (Panel) provided the results of its independent review of the DPO to the Director of RES. The Director of RES agreed with the Panel's conclusions and recommendations which identified improvements to regulatory guidance to support review consistency.

2.9.2. Associated Regulatory Requirements

As previously noted, applicable requirements are found in the following regulations:

- 10 CFR 50.34(a)(1)(ii)(D) which requires an evaluation engineered safety features and barrier that mitigate the release of fission products to the environment
- 10 CFR 50.67 which requires licensees who adopt the alternate source term to evaluate the offsite and onsite dose consequences of a fission product release
- 10 CFR 50, Appendix A, GDC 19, which requires that adequate radiation protection features be provided to permit access and occupancy of the control room
- Various sections of 10 CFR 52, including 10 CFR 52.17(a)(1)(ix), §52.47(a)(2)(iv), §52.79(a)(1)(vi), §52.137(a)(2(iv), and §52.157(d).

2.9.3. Associated Guidance Documents

Several guidance and technical reference documents are used to support DBA radiological consequence reviews:

[7] For example, a typical assumption would be to defeat all ECCS for a two hour period following a LOCA in order to support the fission product source term. However, the underlying regulation does not require safety features to be intentionally defeated for the purposes of performing the analysis, only that the consequences of a fission product source term be evaluated. In other words, it is not necessary to make assumptions that create the source term when performing the assessment.

- Regulatory guide 1.183 describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in complying with regulations for design basis accident (DBA) dose consequence analysis using an alternative source term (AST). This guidance for light-water reactor (LWR) designs includes the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals.

- NUREG-1465, ”Accident Source Terms for Light-Water Nuclear Power Plants,” providing more realistic estimates of the "source term" release into containment than the previous TID-14844, in terms of timing, nuclide types, quantities, and chemical form, given a severe core melt accident. This revised "source term" is to be applied to the design of future Light Water Reactors (LWRs). This NUREG supports the technical basis for information contained in RG 1.183.

- SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” provides guidance to the NRC staff for reviewing radiological consequence analyses for LWRs using ASTs.

- SRP Section 15.0.3, “Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors,” provides guidance to the NRC staff for reviewing radiological consequence analyses for new LWR applications, including advanced evolutionary and passive LWRs.

2.9.4. Potential Implementation Options

At the current time, staff is evaluating and implementing the recommendations arising from DPO-2021-001 and DPO 2020-002. It is expected that the actions arising from these DPO’s will increase the coherence and consistency of guidance associated with design basis accident radiological consequence evaluations. Therefore, the RAAM team did not evaluate additional options at this time, but instead supports continued staff efforts to resolve the previously identified issues.

2.9.5. Potential Impact for Operating LWRs

Increased coherence and consistency in the guidance associated with design basis accident radiological evaluations are expected to have limited impact on operating LWRs. However, plants with limited margin to offsite or control room dose limits may benefit form increased flexibilities when utilizing higher burnup fuel, including accident tolerant fuel, or when changing parameters that impact offsite releases (e.g., containment leakage rates).

2.9.6. Potential Impact for New LWRs

Increasing the coherence and technical basis for guidance associated with design basis accident radiological consequence analysis is expected to provide a significant benefit to new

LWR applicants. Clarity in the desired regulatory basis would help both applicants and staff navigate new or emerging issues and arrive at regulatory decisions that provide flexibility while ensuring the underlying intent of the regulations are met.

2.9.7. -RAAM WG Assessment

The WG believes that actions to resolve recommendations associated with DPO 2020-002 and 2021-001 should continue with high priority. The need for additional work can be reexamined once these actions are complete.

2.9.8. Recommendation

The RAAM Working Group recommends that the staff actions to address recommendations associated with DPO 2020-002 and 2021-001 continue. This issue could be revisited once these actions are completed.

2.10. LEVERAGING LEVEL 3 PRA MODEL RESULTS WITH THE LICENSING MODERNIZATION PROJECT TO FOCUS STAFF REVIEWS

2.10.1. Description

This option would use risk-insights from the NRC's Level 3 PRA project to develop enhanced plant-specific risk-insights similar to Level 3 PRA insights. These enhanced risk-insights will be calculated from a combination of readily available safety correlated factors (i.e., CDF, accident time, plant states, etc). Level 1 PRA models treat all core damage events equally even though they do not represent equal consequences to the public. The large early release frequency (LERF) metric includes additional information beyond the Level 1 PRA model to address this. However, a Level 3 PRA model provides a more complete understanding of plant risks compared to Level 1/LERF PRA models but require extensive amounts of resources to develop. Given the knowledge gained through the NRC's Level 3 PRA project and the tools developed in the LMP framework, there is an opportunity to generate enhanced risk-insights that go beyond the current PRA model capabilities. This would provide both the NRC staff and the nuclear industry with a more accurate representation of the overall plant risks. This approach would require a thorough review of the current risk-metrics used in regulatory decisions and a determination on the benefits provided by using an enhanced risk metric.

2.10.2. Associated Regulatory Requirements

10 CFR 50.34(a)(1)(ii) requires a description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

10 CFR 50.34(f)(1)(i) requires a plant/site specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.D.8)

10 CFR 50.71(h) (1) No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for

initial loading of fuel.

(2) Each holder of a combined license shall maintain and upgrade the PRA required by paragraph (h)(1) of this section. The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations under § 52.110(a) of this chapter.

(3) Each holder of a combined license shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by paragraph (h)(1) of this section to cover all modes and all initiating events.

2.10.3. Associated Guidance Documents

NUREG-0800, Chapter 19, provides staff review guidance for the design specific PRA for design certification (DC) and plant-specific PRA for a combined license (COL).

RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” provides an approach for determining the technical adequacy of PRA results for risk-informed activities.

RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” provides an approach for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights. It provides general guidance concerning analysis of the risk associated with proposed changes in plant design and operation.

NEI 18-04 “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. This guidance document provides one acceptable means for addressing the aforementioned topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection.

RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors,” provides the U.S. Nuclear Regulatory Commission (NRC) staff’s guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-light-water reactors (non-LWRs), including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors at different thermal capacities.

2.10.4. Potential Implementation Options

The main objective of this item is to develop a set of high level general guidelines that can be used to identify risk insights for a variety of applications. Experience with the application of the LMP process to an LWR plant, in addition to past experience with the Enhanced Safety Focused Review Approach (ESFRA), could be used to develop a set of overarching guidelines that can be used to appropriately grade the staff’s review based on risk-informed principles. It is anticipated that this overarching guidelines can be applied on a design specific basis to support detailed staff reviews as appropriate. To implement this approach new guidance will need to be developed, and/or existing guidance will need to be revised. These revisions would be focused on describing high level guidelines for identifying risk significant features of a licensing review and potentially using enhanced risk-metrics with appropriate acceptance thresholds as an alternative to the risk metrics that are currently used.. A technical basis document would need to be developed to establish the technical basis of the high level guidelines and enhanced risk-metric and the corresponding acceptance thresholds for certain applications. This process could be informed by prior work on the ESFRA used by NRO.

2.10.5. Potential Impact for Operating LWRs

The potential impact for operating LWRs is the enhanced risk perspective (e.g., severe accident and consequence insights) that would be available for use and would provide a more accurate representation of plant risks. This may provide an alternative perspective that could be used to enhance established processes. It is however recognized that staff already possesses significant tools for operating plant risk insights, including insights from licensee PRAs and the plant-specific SPAR models. However, the LMP insights provide additional core damage consequence and accident success path insights that can further refine risk approaches. Additional risk insights from LMP experience could be incorporated into the LIC-206, “Integrated Risk-Informed Decision-Making for Licensing Reviews,” process.

2.10.6. Potential Impact for New LWRs

The potential impacts for new LWRs are similar to the operating LWRs in the enhanced risk perspective and more accurate representation of plant risks, but may be more useful since the designs benefits may be more readily seen through the enhanced risk perspective. The benefits to the new LWRs would be greater than for the operating LWRs and would drive for a quicker acceptance of these alternative approaches.

2.10.7. RAAM WG Assessment

The WG believes that this item would provide a low to medium benefit by better focusing staff review resources and could be accomplished with a relatively low resource cost.

2.10.8. Recommendation

Although the working group believes that this item would provide a relatively low benefits for operating reactors and a medium benefit for new reactors, the low implementation cost makes this option worthy of pursuit. The working group believes that leveraging past programs like the enhanced Safety focused Review Approach (ESFRA) used in NRO could enable relatively quick implementation of this option.

2.11. USE OF EVENT SEQUENCE FREQUENCIES TO RISK-INFORM DESIGN BASIS EVENT CATEGORIZATION

2.11.1. Description

In general, regulations use terms such as “anticipated operational occurrence”, “postulated accidents” and “beyond design basis accidents” to categorize postulated operational events. In general, the usage of these terms includes qualitative consideration of their expected frequency of occurrence, but the historical practice has not been to bin events according to a numeric frequency. However, the increased use of PRA techniques (including consideration of uncertainties) offers an approach that may provide more consistency in categorizing these events. For example, NEI 18-04 presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. This guidance document provides one acceptable means for addressing the aforementioned topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection. NRC endorsed NEI 18-04 using RG 1.233 for use by non-LWR applicants applying for permits, licenses, certifications, and approvals under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities” (Ref. 1), and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 2).

The LMP framework uses event sequence frequencies to identify the LBEs. Some, new LWR applicants (e.g., BWRX-300) and the American Nuclear Society have also appears to have proposed using event sequence frequencies to define the DBE and use demarcation of those frequencies to identify safety-related systems. Unlike LMP, these approaches do not rely on the development of all-modes all-sources Level 3 PRA to develop the design and licensing basis. Consequently, these approaches reliance on PRA is consistent with current maturity of the PRA technology. Rather, these approaches have relied heavily on DID to develop technical robust safety strategies.

Under this initiative, insights gained from the recently completed RES application of the LMP to LWRs; results of RES's Level 3 PRA study; and planned review of ANS 30.3 and BWRX-300 (proprietary) could be used to explore methods to bin operational events into appropriate design basis event categories. New reactor developers and applicants could potentially use this approach to support their applications. In addition, recent staff activities associated with risk-informed treatment of emergency preparedness for small module reactors (e.g., NuScale) could be leveraged for pertinent insights.

If endorsed, implementation of this approach is likely to require additional research activities to address known challenges such as processes to identify accident sequences in a predictable and repeatable manner, treatment of uncertainties, and reconciliation of risk-informed binning with traditional deterministic approached. This additional work could be performed under a work request with RES/DRA.

2.11.2. Associated Regulatory Requirements

10 CFR 50.2 defines safety-related structures, systems and components as those structures, systems and components that are relied upon to remain functional during and following design basis events to assure: (1) The integrity of the reactor coolant pressure boundary, (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

For the special case of AOOs (a subcategory of design basis events), GDC 10 specifies that Specified Acceptable Fuel Design Limits (SAFDLs) are defined which are not to be exceeded during any condition of normal operation, including AOOs. Effectively, this approach means that the SAFDLs are set such that no radiological release is predicted for normal operation and all AOOs, which is a more limiting than the guideline exposures referenced in 10 CFR 50.2.

2.11.3. Associated Guidance Documents

Chapter 15.0.1 of NUREG-0800 states that the reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating AOO and postulated accident conditions and has included the effects of single failures in those systems and components. In the event that applicants or licensees propose an exception to allow credit for the operation of certain non-safety-related equipment, the reviewer should review the technical justification and ensure the safety analysis does not credit the non-safety-related SSC as a frontline mitigating system or as being within the primary success path necessary for satisfying the acceptance criteria in Table 15.0-1

NEI 18-04 presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. This guidance document provides one acceptable means for addressing the aforementioned topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection.

RG 1.233 provides the U.S. Nuclear Regulatory Commission (NRC) staff's guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-light-water reactors (non-LWRs), including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors at different thermal capacities.

2.11.4. Potential Implementation Options

It is anticipated that this option could be implemented through enhancements to staff guidance, including the SRP, to establish a predictable, consistent, and repeatable framework for binning design basis events into appropriate categories based on their event sequence frequency and consequences. Additional research will be needed to evaluate insights gained from implementation of the LMP process, the Level 3PRA project, and other ongoing staff review activities to develop a structured framework for implementation.

2.11.5. Potential Impact for Operating LWRs

It is anticipated that this effort would have limited benefit to operating LWRs since the licensing basis event categories have been well established and there would be limited benefit to reclassifying these events (with the possible exception of large break LOCA, which is addressed under a separate RAAM item).

2.11.6. Potential Impact for New LWRs

Based on recent preapplication and review activities, the implications for new light water reactors could be significant. Once a predictable framework is established, new LWR applicants could leverage their PRA results to support binning of licensing basis events into appropriate categories (AOOs, postulated accidents, beyond design basis events, etc).

2.11.7. <u>RAAM WG Assessment</u>

The WG believes that this option has limited benefit for the operating fleet but could have a high benefit to new LWR applicants. The resource cost associated with developing a predictable framework that appropriately considers uncertainties and potential consequences is expected to be medium to high. Further, additional work and stakeholder engagement would likely be needed to support development of this framework.

2.11.8. <u>Recommendation</u>

The working group recommends that this option be pursued. This option is expected to have the largest impact for new reactor licensing but will also have a medium to high resource impact in order to support broad internal and external stakeholder engagement and develop any necessary tools or processes to support broad implementation.

3.0 ANALYSIS

The discussions contained in Section 2 are summarized in the following table. In assessing each the resource impact, benefits, and priority, the working group considered the following attributes:

- Resource impact - staff level of effort (e.g., guidance updates would require fewer resources than a rulemaking), level of needed stakeholder engagement and socialization with staff, amount of existing work that can be leveraged to address issue.
- Benefit – increase in regulatory stability and clarity, licensee/applicant interest and expected demand for licensing actions the associated area, and potential resource savings and efficiency increases.
- RAAM WG Priority - Considered balance of cost/benefit, other related initiatives, and need for new research activities.

Item	Report Section	Resource Impact	Potential Benefit		RAAM WG Priority	Comment
			Operating Reactors	New Reactors		
Redefine Acceptable Fuel Design Limits in GDC 10 (SARRDLs / SAFDLs)	2.1	High	Low	Medium	Additional Study Needed (e.g., Public Meeting). Potentially useful long term	Will require significant engagement
Risk-Informed Guidance for Crediting Non-Safety-Related SSCs	2.2	Medium / High	Low	High	Pursue – High Priority	Will require significant engagement
Use of an Alternate Criteria to 95/95 for LOCA	2.3	Low	Medium	Medium	Pursue – High Priority	Could potentially be done quickly
Redefine Large Break LOCA to Beyond Design Basis Event	2.4	High	High	High	Pursue – Under Increased Enrichment Rulemaking	Consistent with IE Schedule
Reconsideration of LOCA Break Locations	2.5	Medium / High	Low	Low	Defer – potentially limited interest from industry	-
Risk-inform Single Active Failure Criteria	2.6	Medium	Low	High	Pursue	Will require significant engagement
Define Single Passive Failures for Fluid Systems	2.7	Medium / High	Low	High	Pursue	Will require significant engagement
Risk-Inform EQ Radiological Requirements	2.8	Medium	Low	Low	Defer	-
Increase the Coherence and Consistency of DBA Radiological Consequence Analysis	2.9	-	-	-	Continue to pursue resolution of DPO 2020-002 and 2021-001 with high priority	-

Item	Report Section	Resource Impact	Potential Benefit		RAAM WG Priority	Comment
			Operating Reactors	New Reactors		
Use of LMP Results to Focus Staff Reviews	2.10	Low	Low	Medium	Pursue	Can be implemented in a manner similar to ESFRA. Could be done relatively quickly.
Use of Event Sequence Frequencies to Risk-Inform Design Basis Event Categorization	2.11	Medium / High	Low	High	Pursue	Will likely require additional work and stakeholder engagement to develop framework

In assessing these various options, the RAAM WG noted that the expected benefits for operating reactors tends to be lower than new reactors. This is attributed to several factors including: (1) the development and refinement of risk informed licensing processes for the operating fleet for over 30 years; (2) implementation of many innovative initiatives over the last 30 years, including RG 1.174, PRA standards (and RG 1.200), specific applications including AOT extensions, RI surveillance intervals and outage times, NFPA 805, risk informed ISI, Maintenance Rule, 50.69; and (3) the high number of operating reactor licensees have taken advantage of these processes and obtained significant flexibilities and efficiencies. However, the RAAM WG notes that there are still opportunities for further enhancements for the operating fleet.

4.0 CONCLUSIONS

As directed by the RAAM charter, the RAAM WG investigated the accident analysis methods as presented in SRP Ch. 15. Based on a review of the guidance in conjunction with applicable regulations, recent Commission direction, and advances in the use of risk insights, the RAAM WG has identified a number of review areas that could be modernized to simplify the accident analysis approach while preserving the intended safety intent of the associated regulations. The RAAM WG has documented a preliminary analysis of the level of staff effort which would be required to implement each modernization effort and the likely level of interest for both operating reactor licensees and new reactor applicants. Based on this preliminary analysis, the RAAM WG documented their recommendation to either proceed or not for each of the identified modernization efforts.