U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

DRAFT DRA-ISG-2021-01

Supplemental Guidance for Radiological Consequence Analyses Using Alternative Source Terms

DRAFT Interim Staff Guidance

Month 2021

Note: This draft interim staff guidance is being made public to support the November 2, 2021 Advisory Committee on Reactor Safeguards Full Committee Meeting. This document has not been finalized as an NRC agency position and is therefore subject to changes.

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DRAFT INTERIM STAFF GUIDANCE

Supplemental Guidance for Radiological Consequence Analyses Using Alternative Source Terms

DRAFT DRA-ISG-2021-01

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) staff is providing this interim staff guidance (ISG) on the presence of the power conversion system (PCS) and its ability to provide a large holdup and retention volume for leakage from the main steam isolation valve (MSIV). This ISG will help to resolve differences between the licensee's methods and assumptions and those deemed acceptable to the NRC staff when reviewing license amendment requests (LARs) that propose an increase in the MSIV leakage allowed by technical specifications (TS) for boiling water reactors (BWRs). The staff should acknowledge the presence of the PCS and its ability to provide a large holdup and retention volume for MSIV leakage when staff determines that the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations.

BACKGROUND

This ISG is intended to provide guidance for the NRC staff reviewing LARs asking to increase the MSIV leakage allowed by TS at BWRs. This ISG is not intended as standalone guidance but instead supplements NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Agencywide Document Access Management System (ADAMS) Accession No. ML003734190).

The staff evaluated whether modern analysis approaches and operating experience could be used to inform the reviews of the MSIV leakage increase LARs. The staff approved General Electric Company (GE) Topical Report NEDC-31858P, Revision 2, "BWROG [Boiling Water Reactor Owners Group] Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," issued September 1993 (ADAMS Accession No. ML993440253, not publicly available), with a safety evaluation dated March 3, 1999. Since that time, the staff has developed additional information regarding the use of risk information, seismic hazards, and operating experience following seismic events and severe accidents. The staff's evaluation considered Commission direction on risk-informed and performance-based regulation (e.g., Staff Requirements Memorandum (SRM)-SECY-98-144 at ADAMS Accession No. ML003753601), SRM-SECY-19-0036, "Staff Requirements—SECY-19-0036—Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," dated July 2, 2019 (ADAMS Accession No. ML19183A408) and feedback from stakeholders. In SRM-SECY-19-0036, the Commission stated, "[i]n any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the single failure criterion is unnecessary to provide for reasonable assurance of adequate protection of public health and safety."

As noted in an NRC memorandum, "Implementing Commission Direction on Applying Risk-informed Principles in Regulatory Decision Making," dated November 19, 2019 (ADAMS Accession No. ML19319C832), the staff's application of risk-informed decision making continues to evolve as improved realism, evaluation techniques, and additional information are applied to improve regulatory decision making. The development of the ISG serves as an example of NRC's continuous efforts in working toward being a more modern and risk-informed regulator.

RATIONALE

In 2019, licensees submitted multiple LARs requesting an increase in the MSIV leakage allowed by TS for BWRs. Most BWR licensees previously received approval of their MSIV leakage limit as part of their alternative source term (AST) LAR pursuant to 10 CFR 50.67, "Accident source term." Those AST LARs, which were submitted prior to 2010, typically included consequence analyses for a postulated maximum hypothetical accident. The analyses were based on the assumption that the plant would experience (1) a substantial core melt with subsequent release of appreciable quantities of fission products into the drywell and (2) release of the diluted fission products at the maximum MSIV leak rate allowed by the TS. These accident analyses were intentionally conservative to compensate for known uncertainties in accident progression.

The deterministic approach of the licensees' dose calculation of the MSIV leakage pathway typically credits only safety-related or seismic Category I structures, systems, and components (SSCs) to mitigate the radiological consequences of the accident. PCSs, including the main steam piping downstream of the outboard MSIV and the main condenser, typically are not safety related or considered a seismic Category I SSC. These deterministic analyses assume those SSCs are unavailable and that all or most of the MSIV leakage travels directly to the atmosphere beyond the outboard MSIV.

In 1999, the NRC staff approved a method using the main steam drain lines to direct the MSIV leakage to the main condenser as an alternate pathway to demonstrate compliance with the regulations without relying on only safety-related or seismic Category I SSCs to mitigate the radiological consequences of a postulated release. Specifically, in 1993, GE submitted a topical report, NEDC-31858P, Revision 2, for review by the NRC staff. NEDC-31858P used earthquake experience data, primarily from nonnuclear facilities, to demonstrate the availability of an alternate pathway through the main steam drain lines and the condenser at a plant's safe-shutdown earthquake and, consequently, to justify credit for the pathway in deterministic dose calculations. The NRC staff approved the use of the alternate pathway using the approach in NEDC-31858P, subject to certain limitations in its safety evaluation for the approach. Since that time, approximately 50 percent of the 30 plants that submitted their AST LARs before 2010 used that approach and were able to credit certain SSCs in the PCS to mitigate the radiological consequences. Those licensees were required to provide plant-specific information to address the limitations in the safety evaluation.

The alternate pathway discussed above, particularly the condenser, provides a large holdup volume for fission products and time for physical processes that reduce the release of fission products to the environment. This change in fission product release results in a reduction in the calculated dose. None of the 2019 LARs proposed to credit these pathways for holdup. The staff learned that the resources needed to obtain the plant-specific information to support the staff's determination that the credited SSCs are seismically robust contributed to the licensees' decision not to apply for credit for the alternate pathway.

Consistent with previous Commission direction on risk-informed and performance-based regulation (e.g., SRM-SECY-98-144 and SRM-SECY-19-0036), and considering feedback from stakeholders, the staff evaluated whether current analysis approaches, data, and operating experience gained since the approval of NEDC-31858P could be used to inform the reviews of the 2019 LARs, without the need to obtain the plant-specific information. Subsequently, the staff developed a technical assessment (Appendix A to this ISG) to identify an important source of realism that can be used by the staff to inform its reviews.

In its technical assessment, the staff identified the PCS as a realistic and available hold-up volume for fission products. The staff further evaluated the seismic capacity of the SSCs in the PCS, including the main steam piping, equalization header, and condenser, to determine whether these SSCs would be available to provide a hold-up volume for fission products following a safe shutdown earthquake (SSE). The staff used engineering information, such as operations and design knowledge, as well as probabilistic and risk information, in its assessment. The staff also leveraged recent relevant operating experience, such as that obtained from the Fukushima Daiichi accident and the earthquake that affected the North Anna Power Station.

The staff's technical assessment concluded that there is high confidence in the ability of the SSCs in the PCS to provide a volume for hold-up and retention of fission products. Further, the assessment concluded that the probability that the PCS would be unavailable to serve as a volume for hold-up and retention at an SSE is low. These conclusions provide useful insights and guidance to the staff for decision-making on reviews of MSIV leakage increase LARs. Specifically, the high probability that doses will be lower than those estimated strictly using traditional deterministic methods, which include accepted assumptions that do not credit hold-up and retention of the MSIV leakage within the PCS, can be used by the staff as part of the information for its reasonable assurance finding. This ISG will not change the acceptable methods used by the licensee to demonstrate conformance with 10 CFR 50.67.

Since the NRC staff has developed a technical assessment for the updated guidance in SRP 15.0.1 to reflect current technical knowledge, including operating and risk insights, new staff guidance is warranted. This interim guidance is needed prior to the next update of the SRP 15.0.1 to support its use by staff for MSIV leakage increase LAR reviews, and to inform external stakeholders about the updated staff guidance. In addition, the issuance of the proposed ISG will facilitate receipt of comments from external stakeholders which can expedite the inclusion of the new guidance into the SRP, as applicable.

APPLICABILITY

All holders of an operating license or construction permit for a nuclear power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of a combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants."

GUIDANCE

This ISG provides supplemental guidance to items III.6.c and IV.5 in SRP Section 15.0.1. The basis for the supplemental guidance is a technical assessment that uses knowledge and operating experience related to the PCS, including information on the seismic capacity and risk at nuclear power plants. Appendix A to this ISG details the technical assessment supporting the supplemental guidance.

IMPLEMENTATION

The staff, through this ISG, should acknowledge the presence of the PCS and its ability to provide a large holdup and retention volume for MSIV leakage when staff determines that the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations. In doing so, the staff should recognize that there is a high probability that doses will be lower than those estimated using deterministic methods that include accepted assumptions but do not credit holdup and retention of the MSIV leakage within the PCS. The staff can use acknowledgement of the presence of the PCS as part of the information for its reasonable assurance finding. This ISG does not change the acceptable methods used by the licensee to demonstrate conformance with 10 CFR 50.67, “Accident source term,” and is consistent with the Commission directions in SRM-SECY-98-144 and SRM-SECY-19-0036.

In a future SRP update, the staff plans to incorporate similar language in item III.6.c in SRP Section 15.0.1 to incorporate this ISG into guidance.

Through the use of this ISG, the staff may use the following concluding paragraph in their safety evaluations, if appropriate:

> The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed license amendment. The NRC staff finds the analysis methods and assumptions consistent with the applicable regulatory requirements and guidance. The NRC staff concludes with reasonable assurance, based in part on the risk and engineering insights to compensate for uncertainties in the evaluation of the dose consequences from the MSIV release pathway, that the licensee’s dose estimates will comply with the acceptance criteria.

In a future SRP update, the staff plans to add the above paragraph to item IV.5 in SRP Section 15.0.1 to incorporate this ISG into guidance.

BACKFITTING AND ISSUE FINALITY DISCUSSION

The guidance in this ISG-2021-01 clarifies how the NRC staff will acknowledge the presence of the PCS and its ability to provide a large holdup and retention volume for MSIV leakage when staff determines that the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations. Issuance of this ISG does not constitute backfitting as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109 (the Backfit Rule) and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting,

Issue Finality, and Information Requests"; would not affect the issue finality of an approval issued under 10 CFR Part 52; and would not constitute forward fitting as that term is defined and described in Management Directive 8.4.

Further, the NRC staff does not, at this time, intend to impose the positions represented in the ISG in a manner that would constitute backfitting or forward fitting or affect the issue finality of a Part 52 approval. If, in the future, the staff seeks to impose a position in the ISG in a manner that constitutes backfitting or forward fitting or does not provide issue finality as described in the applicable issue finality provision, then the staff would need to address the Backfit Rule, the forward fitting criteria in Management Directive 8.4, or the applicable issue finality criteria, respectively.

CONGRESSIONAL REVIEW ACT

Discussion to be provided in the final ISG.

FINAL RESOLUTION

By September 2022, this guidance will be transitioned into SRP Section 15.0.1 in conjunction with a separate ongoing effort to revise Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792) expected to be completed by July 2022. In addition to this guidance, SRP Section 15.0.1 will also include a reference to the revised RG. 1.183. Following the transition of this guidance to the SRP, this ISG will be closed.

APPENDICES

A. Technical Assessment Supporting the Interim Staff Guidance
B. References
C. Public Comment Resolution

APPENDIX A

Technical Assessment Supporting the Interim Staff Guidance

This technical assessment provides the basis for DRA-ISG-2021-01 related to the U.S. Nuclear Regulatory Commission (NRC) staff's review of the radiological consequences of leakage from a boiling-water reactor (BWR) main steam isolation valve (MSIV) during a postulated maximum hypothetical accident (MHA) involving significant core damage, which is typically assumed to occur in conjunction with a large loss-of-coolant accident (LOCA).[1] The staff evaluated the ability of a realistic transport pathway through the structures, systems, and components (SSCs) in the power conversion system (PCS), including the main steam line (MSL) piping and equalization header, to provide large holdup volume for fission products (primarily aerosols). This technical assessment is a structured evaluation of the acceptability of dose consequence analyses for MSIV leakage for staff use when the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations.

The deterministic dose calculations for MSIV leakage using the MHA were not intended to represent actual event sequences. Instead, they were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression.

The deterministic dose calculation of the MSIV leakage pathway typically credits only safety-related or seismic Category I SSCs to mitigate the radiological consequences and to estimate conservative doses. The PCS, including the main steam piping downstream of the outboard MSIV and the main condenser, typically is not safety-related or considered a seismic Category I SSC. Consequently, these deterministic dose calculations assume those SSCs are unavailable and all or most of the MSIV leakage travels directly to the atmosphere beyond the outboard MSIV. However, a realistic consideration of the typical configuration of a BWR main steam system provides holdup volumes for fission product retention and decay resulting in significantly reduced releases.

The NRC staff has previously approved alternative methods for showing compliance with the regulations for the MSIV leakage pathway. In 1999, the NRC staff approved credit for an alternate pathway through the main steam drain lines and the condenser using the approach discussed in NEDC-31858P, Revision 2, "BWROG [Boiling Water Reactor Owners Group] Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," issued September 1993 (Reference 1), subject to the limitations in its safety evaluation (SE) dated March 3, 1999 (Reference 2). The credit for the alternate pathway in the dose calculations is provided through a model that considers the hold-up, dilution, and deposition. The alternate pathway, especially the condenser, provides a large holdup volume for fission products. This results in a reduction in the rate of fission product release in the deterministic dose calculations and a reduction in the calculated dose. The SE on NEDC-31858P, Revision 2, gives precedent for not relying on only safety-related or seismic Category I SSCs for mitigating the radiological consequences of a postulated release. That SE states that requiring

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[1] For simplicity, the remainder of this evaluation will use the term "maximum hypothetical accident" (MHA) for such a postulated accident.

the nonseismically analyzed portions of the main steam system piping and components to meet seismic Category I requirements is impractical because the modifications required to upgrade the system to seismic Category I requirements would be very costly.

In addition, the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 3), allows credit for the condenser, which is a nonsafety-related SSC, without any additional information or analysis from the licensee related to the "seismic robustness" at a plant's SSE for the deterministic dose analysis for the rod drop accident. However, the same RG does not credit the condenser without further analysis for "seismic robustness" at a plant's SSE for the deterministic dose analysis for the MHA. The reason for the differing treatment of the same SSC under the same seismic loading (i.e., the plant's SSE) is unclear.

This assessment considers fission product transport through the PCS pathway, including the MSLs and steam equalization header, rather than a direct release to the atmosphere as is usually postulated in the deterministic dose calculations. In other words, the assessment addresses the risk of fission products not transporting through the PCS pathway. This assessment uses engineering information, such as operations and design knowledge, and probabilistic and risk information on the seismic capacity (i.e., the ability of an SSC to withstand acceleration induced by a seismic event) of the SSCs in the realistic transport pathway to determine the risk of unavailability of the SSCs in the PCS pathway for fission product holdup and retention. Figure 1 shows the assessment approach, discussed further in Section 2. Section 2.1 discusses the likelihood of a realistic pathway not being available. Sections 2.3 and 2.4, respectively, discuss the failure probability of the SSCs in the realistic pathway at a plant's safe-shutdown earthquake (SSE) and the frequency of an undesired outcome (radiological release). Section 2.5 discusses the uncertainty evaluation.

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Based on the assessment summarized in this document, the staff concludes that the risk of the unavailability of SSCs in the realistic transport pathway through the SSCs in the PCS, including the MSL piping and the steam equalization header, for fission product holdup and retention is low, including at seismic accelerations corresponding to a plant's SSE. In addition, further conservatisms in this assessment provide additional defense in depth and maintain the safety margin. Therefore, in evaluating the acceptability of dose consequence analyses for MSIV leakage when the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations, it is acceptable for the staff to consider risk insights regarding the availability of the PCS in reaching its reasonable assurance finding.

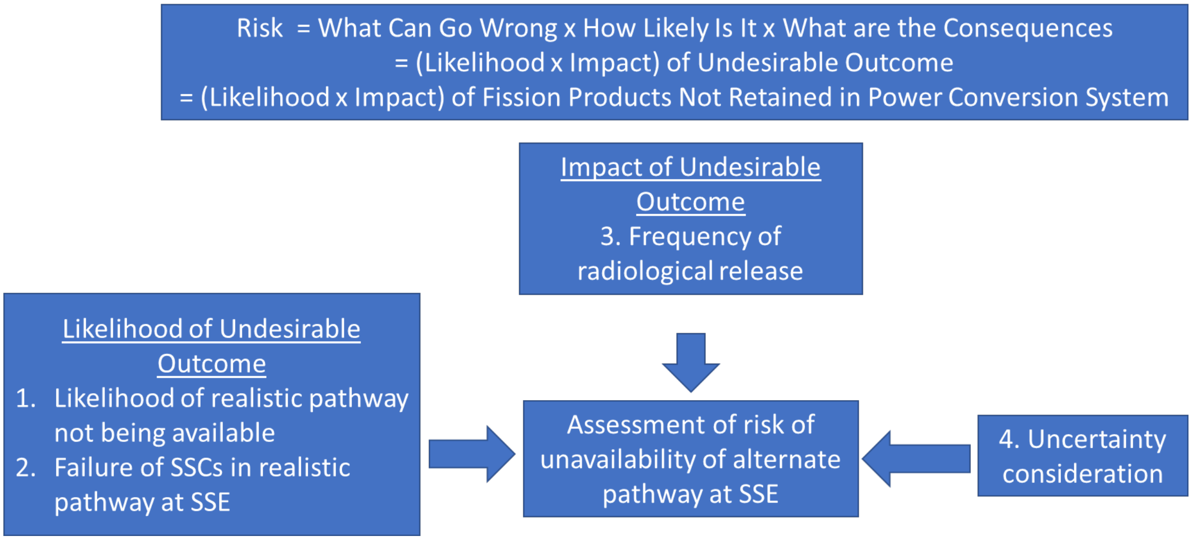

Figure 1. Assessment approach

1. Background

1.1. Regulatory Requirements

Each application for a construction permit is required to include a safety assessment of the facility site in the safety analysis report that addresses the site evaluation factors included in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor site criteria," (Reference 4) including analysis and evaluation of major SSCs that bear significantly on the site assessment. These evaluation factors include the characteristics of the reactor, use and population characteristics of the site environs, and the physical characteristics of the site.

As an aid to evaluating the site for applications dated before January 10, 1997, 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," specifies an analysis of offsite doses that considers an assumed fission product release from the core (i.e., source term) not exceeded by any credible accident, the expected demonstrable leak rate from containment, and the meteorological conditions pertinent to the site. In addition, 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants" (Reference 5), General Design Criterion 19, "Control room" (or a similar principal design criterion), also applies. Under 10 CFR 50.67, "Accident source term," a licensee that seeks to revise its source term used in design-basis radiological consequence analyses must reevaluate the consequences of applicable design-basis accidents previously analyzed in the safety analysis report. The NRC may issue a license amendment adopting the revised source term "only if the *applicant's analysis* demonstrates with *reasonable assurance*" (emphasis added) that the dose criteria specified in 10 CFR 50.67(b)(2) are not exceeded for the exclusion area boundary, the low population zone boundary, and the control room for any accident considered credible.

1.2. System Information

The MSIVs installed on the MSLs in BWRs isolate the reactor system in the event of a break in a steam line outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Each MSL has two MSIVs: the inboard and outboard MSIV. The

outboard MSIVs form the outermost part of the reactor coolant pressure boundary along the MSLs.

From the MSIVs, the main steam system transports steam to the main turbine, the turbine bypass valves, and various auxiliary equipment. Typically, the main steam system consists of four large-diameter MSLs from each outboard MSIV to a large-volume main steam equalizing header. Some BWR facilities have a third motor-operated isolation valve in each steam header between the outboard MSIV and the equalizing header. From the equalizing header, steam is typically supplied to the following components:

- four turbine stop valves (TSVs) and four control valves in series through large-diameter steam lines
- two turbine bypass valves that discharge steam directly to the main condenser through diffusers
- main feedwater pump, when steam-turbine driven and not electric (at startup and low power; may switch to extraction steam at high power)
- moisture separator-reheaters
- high-pressure feedwater heaters

In addition, the MSLs are equipped with drain lines from low points in the piping that included a steam trap and parallel motor-operated isolation valve to direct drainage to the main condenser. Drain lines at some facilities have been removed from service.

1.3. Dose Consequence Evaluation

Based on the requirements in 10 CFR 50.34, “Contents of applications; technical information,” licensees performed their MHA analyses to conservatively reflect the various fission product release pathways based on the fission product concentrations of the containment. The fission product releases into containment are used for evaluating the acceptability of both the plant site and the effectiveness of engineered safety feature components and systems. Although the MSIVs are designed to provide a leak-tight barrier, some leakage through the valve seat will occur, and an allowable leakage value is part of a plant’s technical specifications (TS). Based on the assumptions used for the MHA (i.e., following a design-basis LOCA with no credit for nonsafety-related components and assuming the single failure of one MSIV to close), the design-basis maximum allowable leakage through the MSIVs would be the numerical value presented in the TS. As mentioned, this limit on MSIV leakage is to maintain offsite and control room radiological consequences to within the regulatory limits in the event of an accident. For amendments associated with the revised accident source term at facilities with original operating licenses issued before January 10, 1997, the NRC specifies the accident dose consequence analysis regulatory limits in 10 CFR 50.67(b)(2).

The NRC staff issued regulatory guidance for dose consequence analyses using the revised source term in RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” (Reference 3);[2] Regulatory Issue Summary 2006-04,

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[2] Unless otherwise specified, references to RG 1.183 in this document refer to Revision 0 of RG 1.183, rather than to the proposed Revision 1, which is currently under development by the staff.

"Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (Reference 6); and guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 15.0.1, Revision 0, "Radiological Consequence Analyses Using Alternative Source Terms," issued July 2000 (Reference 7).

This guidance and NRC regulations generally do not credit the capabilities of SSCs beyond the outboard MSIV to mitigate fission product release unless those SSCs can be shown to be "seismically robust." As such, fission product leakage into the MSLs that is neither collected in a leakage control system (LCS)[3] nor retained within the MSL upstream of the outboard MSIV is assumed to go directly to the turbine building. However, consideration of the main steam system, including the MSLs beyond the outboard MSIV and the steam equalization header, results in more realistic pathways for fission product leakage.

In January 1983, the NRC staff initiated Generic Issue (GI) C-8, "MSIV Leakage and Leakage Control Systems Failures," to assess (1) the cause of MSIV failures, (2) the effectiveness of the LCS and alternative leakage paths, and (3) the need for regulatory action to limit public risk. This GI considered the actual natural phenomena associated with the behavior and the characteristics of radioactive materials and the historical capability of nonsafety-related components to survive seismic events. The staff documented the results of its assessments in NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods," published August 1986 (Reference 8). Concurrently, the BWROG formed the MSIV Leakage Control Committee to determine the cause of high leakage rates associated with many of the MSIVs and to develop recommendations for reducing the leakage rates. The committee provided recommendations and comments to the staff in February 1984 and April 1986. In 1990, the NRC published NUREG-1732, "Regulatory Analysis for the Resolution of GI C-8, Main Steam Isolation Valves Leakage and LCS Failure" (Reference 9). NUREG-1732, which was a follow-on regulatory analysis to NUREG-1169, documented the NRC staff's conclusions that no backfit requirements to reduce public risk were warranted and that no regulatory actions should be taken. One of the alternative resolutions of GI C-8 showed that several nonseismic Category I alternate MSIV leakage paths resulted in lower doses.

In 1993, General Electric Company (GE) submitted a topical report, NEDC-31858P, Revision 2, for review by the NRC staff. NEDC-31858P used earthquake experience data, primarily from nonnuclear facilities, to demonstrate the availability of the alternate pathway at a plant's SSE and, consequently, justify credit for the pathway in dose calculations. Figure 2 gives a schematic illustration of the pathway. In the March 3, 1999 SE for NEDC-31858P, Revision 2 (Reference 10), the NRC staff approved the use of the alternate pathway using the approach discussed in NEDC-31858P subject to the limitations in the SE. These limitations include demonstration by the licensee that the alternate pathway would be "seismically robust" at the plant's SSE.

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[3] Originally, many of the BWR designs included MSIV LCSs to collect MSIV leakage and direct it to the standby gas treatment system, where the leakage would be processed and directed to an elevated release point post-accident. However, these systems were designed for relatively low leakage rates, and operators had problems maintaining conservative MSIV leakage rates determined via local leak rate testing within the leakage control system design capability. Therefore, many of the MSIV LCS systems were removed or no longer used.

The NRC staff reviewed past SEs of BWR MSIV leakage dose consequence analyses, encompassing 20 SEs representing 30 individual plants, and determined that slightly over 50 percent (16/30) of the plants took credit for a seismically robust path to the condenser. Consistent with the limitations in the SE for NEDC-31858P, Revision 2, the licensees provided a plant-specific alternate path and the bases for its functional reliability at the corresponding SSE along with a list of manual actions to direct MSIV leakage to the condenser, if needed.

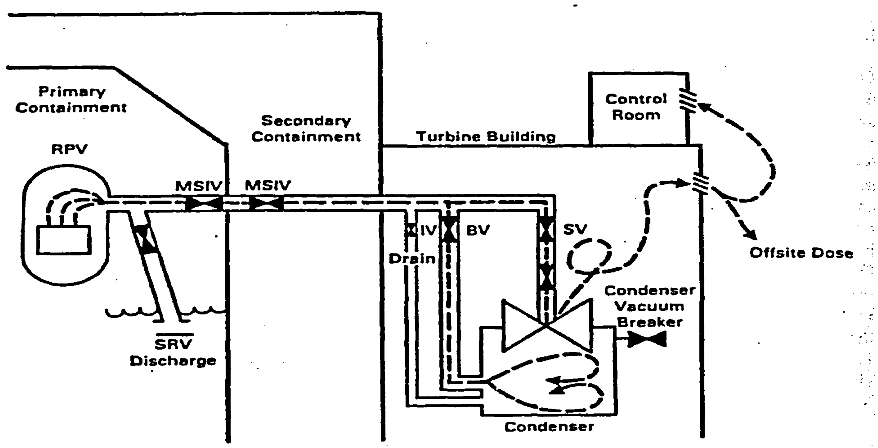

Figure 2. Schematic illustration of the alternate pathway (reproduced from Reference 11)

1.4. Consideration of Fission Product Dilution and Holdup in the Power Conversion System

The NRC staff assessed whether information obtained since the time Revision 2 of NEDC-31858P was approved could be used to support the staff's review of MSIV leakage license amendment requests (LARs). Specifically, the staff evaluated whether a realistic transport pathway through the PCS, including the MSLs downstream of the outboard MSIV and the steam equalization header, can provide holdup volume and can be considered "seismically robust" to support the staff's reasonable assurance finding for the MSIV leakage increase LARs.

This assessment supports the interim staff guidance for the NRC staff's review and reasonable assurance finding for the deterministic dose calculations submitted as part of the proposed increase in the MSIV leakage specified in a plant's TS. Figure 3 illustrates the relationship of this assessment to the BWR MSIV leakage dose consequence analyses.

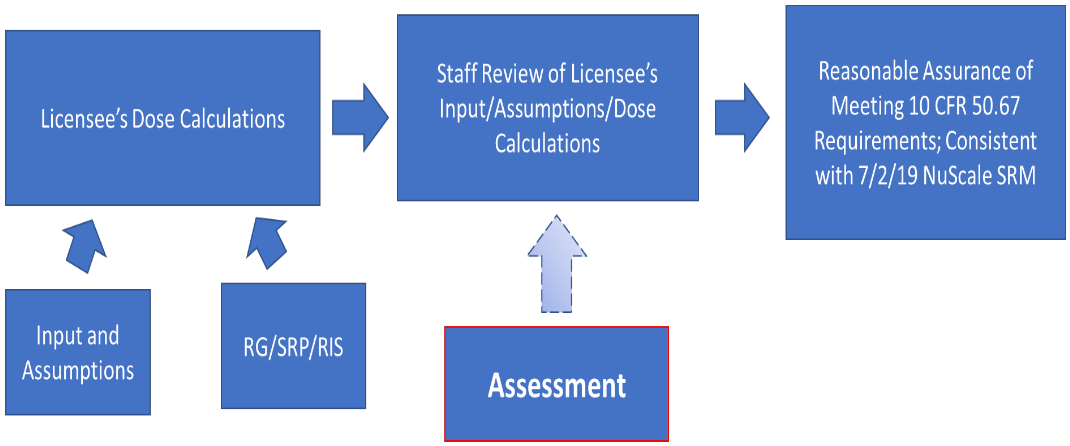

Figure 3. Relationship of this assessment to MSIV leakage dose consequence analyses

2. Detailed Assessment

The staff evaluated the likelihood that the main steam and power conversion systems would serve to effectively mitigate the dose consequences of MSIV leakage. As part of its assessment, the staff used engineering insights as well as probabilistic and risk information related to seismic events.

2.1 Engineering Insights

The postulated scenario considered in the dose calculations is a low-likelihood event involving an MHA with postulated assumptions that include failure of emergency core cooling leading to core damage (note that a large-break LOCA, by itself, has a low occurrence frequency). A previous staff evaluation of MSIV leakage using a probabilistic approach in PRAB-02-01, "Assessment of BWR Main Steam Line Release Consequences," issued October 2002 (Reference 11), determined that the sequence most likely to lead to a large release through the MSIV leakage path would be a short-term station blackout, where both alternating and direct current (AC and DC) electric power sources are lost early in the event. The analysis determined that (1) the short-term station blackout has a very low frequency of occurrence, due, in part, to the highly reliable vital AC and DC electrical distribution systems as well as redundancy in high- and low-pressure core cooling systems and (2) MSIV leakage rate orders of magnitude higher than the typical TS limit would be necessary for this very low frequency event to result in exceeding the dose limits associated with the Commission's Safety Goal Policy Statement (51 FR 30028 (August 21, 1986)).

This sequence is similar to the actual event progression for the accident at the Fukushima Daiichi nuclear power plant in Japan following the Great Tohoku Earthquake of 2011 and the resulting beyond-design-basis tsunami that led to a total loss of AC power and substantial degradation to the DC electrical distribution. However, it is important to note that the radiological consequences of that event were dominated by releases directly from the containment to the reactor building, particularly releases through the drywell head due to above-design internal containment pressure (see Reference 12). Similarly, leakage directly from the containment to the reactor building would be expected to dominate the consequences of other accident sequences, including LOCA sequences with inadequate core cooling, for two

reasons. First, the more probable event sequences leading to fuel damage also degrade the active systems that enhance primary containment heat removal and secondary containment performance. Second, the main steam system downstream of the MSIVs is a high-pressure system with normally low leakage.

Even under the postulated MHA, the main steam system and other components of the PCS would mitigate leakage beyond the MSIVs. The main steam system, including the equalization header, is a high-pressure and high-temperature system with a large internal volume, which offers a large holdup volume for fission products along with the effects of dilution and fission product settling or deposition. The high-pressure and high-temperature design assures margin in material strength to accommodate seismic loads under the low pressure and temperature conditions that would exist based on the postulated post-accident conditions for the MHA. Post-accident conditions would also support condensation of water vapor in the gases leaking from the MSIVs, which would enhance the ability of the main steam system to retain the fission products.

The staff evaluated the strength of the main steam piping downstream of the second MSIV by surveying BWR plants to identify design standards and quality classifications applicable to that piping. In the plants with BWR 3 and BWR 4 designs, this piping is typically designed to American Society of Mechanical Engineers (ASME) Standard B31.1.0, “Power Piping” (Reference 13), or equivalent, and constructed to augmented quality standards in the areas of material certification, testing, and nondestructive examination. In plants with BWR 5 and BWR 6 designs, this piping is typically seismically qualified, designed to ASME Boiler and Pressure Vessel Code, Section III, Class 2 standards for nuclear piping, and treated as safety related. Therefore, the design standards provide additional confidence about the robustness of the main steam piping in the PCS. Any leakage beyond the main steam system piping would encounter additional volumes, such as the steam admission chambers for the high-pressure turbine, that provide additional reduction in fission product release compared to a direct release to the turbine building atmosphere. Thus, even the most direct leakage paths achieve a reduction in fission product release, and engineering insights support the availability of the main steam and power conversion systems to provide the reduction.

The MSIV leakage limit, as tested, includes leakage from the valve stem (i.e., through the valve packing). Only the outboard MSIV packing leaks outside of the primary containment. The leakage through the packing represents a small fraction of the leakage, because such leakage must follow a tortuous path through the packing. Further, the flow area through the packing is small, resulting in a small leak rate because the leak rate is dependent on the flow area. Also, the packing leakage from the outboard MSIV is to the relatively large and structurally robust steam tunnel space. Leakage from the steam tunnel space to the environment would typically be around blowout door seals to the turbine building. Therefore, most of the leakage will be through the MSIV seat, which is addressed in this assessment. This discussion is equally applicable to so-called “other identified leakage.” It should be noted that leakage from the PCS is detrimental to the at-power operation of a plant and is, therefore, expected to be promptly identified and corrected.

Therefore, while containment performance for the MHA is important to defense-in-depth, the current regulatory guidance does not necessarily include appropriate consideration of the robust, passive components downstream of the MSIVs. The MHA is postulated based on 10 CFR 50.67, but available information suggests that conservatism in the disregard of components downstream of the MSIVs can result in over-allocation of resources to improve the

low-pressure sealing of the MSIV seats and actual doses to individuals performing such activities.

2.2 *Realistic Transport Pathway*

The staff considered the reliability of MSIVs to close upon demand (in failure terms, the probability of MSIVs to fail to close on demand) using the 2015 update of the component reliability data sheet used for the failure probabilities in the NRC Standard Plant Analysis Risk models (Reference 14). Based on this information, the mean probability of the failure of an MSIV to close is about $9x10^{-4}$ per demand. Therefore, the MSIVs are highly reliable in closing upon demand. Note that the failure probability is for a single MSIV; therefore, the probability of failure of both the inboard and outboard MSIVs will be lower. Further, the 95th percentile of the probability of failure to close for each MSIV is about $1.2x10^{-3}$ per demand and confirms this conclusion.

Drain lines and high-pressure steam lines to plant auxiliaries (e.g., steam-turbine-driven main feed pumps, moisture separator reheaters, high-pressure feedwater heaters) from the MSLs of BWRs are isolated using motor-operated valves. In parallel, the drain lines may contain an automatic steam trap and orifice that provide for automatic draining of condensate from the steam lines. The 2015 update of the component reliability data sheet provides the mean probability of the failure of a motor-operated valve to close as approximately $3x10^{-4}$ per demand, with the 95th percentile value approximately $8x10^{-4}$ per demand. The staff recognizes that some BWRs have capped the drain lines from the MSLs because the drain lines were not required for startup and shutdown. All components receiving main steam normally return the condensate to the condensate system, whether via a feedwater heater and the heater drain collection system or directly to the main condenser hotwell.

PRAB-02-01 determined that, for the case where the MSIVs, turbine by-pass valves, and drain lines remain closed, the path for the leakage through the MSIVs would be through the TSV and turbine control valve (TCV), then into the turbines (high and low pressure) and turbine steam seals. These valves are routinely tested for turbine overspeed protection purposes, and licensees maintain the governor valves with low seat leakage when closed to preclude excessive turbine speed when the turbine is unloaded. However, their large size and the lack of seating pressure could allow leakage at the MSIV leakage rate to the main high-pressure turbine.

Therefore, based on the available reliability data for components encountered in the release path, the highest probability outcome for fission product transport for deterministic dose calculation is that any MSIV seat leakage would be held up within the large-volume MSLs and the steam equalization header. If leakage passes the TSV and TCV or if random failure of the valves to close is assumed, the main turbine along with other PCS SSCs (such as the main condenser) provide additional holdup volume for fission products.

As noted above, the main condenser provides a large volume for fission product holdup and retention. The large holdup volume in the MSLs beyond the outboard MSIV as well as the steam equalization header would reduce the leakage compared to that from the outboard MSIV. In addition, the flow between the TSV and the high-pressure turbine will be governed by pressure differential; because the pressure differential is small, the flow will be small. For leakage that reaches the main turbine, the turbine blades provide deposition surfaces and the turbine steam seal is a tortuous labyrinth, resulting in further minimizing any fission product release. The high-pressure turbine discharge reaches the low-pressure turbine through the

moisture separator/reheaters and leakage around quick-acting butterfly valves. From the low-pressure turbine, a pathway to the main condenser still exists because of the discharge connection of the low-pressure turbine to the main condenser. Therefore, any leakage from the turbine shaft labyrinth seals will be low compared to that from the outboard MSIV. Formal credit for the holdup in the condenser in the deterministic dose calculations consistent with accepted regulatory positions assumes that the condenser is "open" (i.e., a fixed amount of leakage, specified in RG 1.183, leaves the condenser).

In summary, consideration of engineering insights, available reliability data, and realistic transport pathways for fission products would result in a large holdup volume for fission products. This could support the NRC staff's reasonable assurance finding for its review of the deterministic dose calculations associated with LARs for MSIV leakage increase.

2.3 Reliability of Structures, Systems, and Components in the Realistic Transport Pathway Under Seismic Events

The probability of failure (and, consequently, reliability) of an SSC under seismic demand is represented by the fragility of the SSC. Higher fragility means lower failure probability or higher reliability of that SSC under seismic demand. Seismic fragility values are expressed in terms of multiples of gravitational acceleration (e.g., 0.5g) and, unless otherwise noted, expressed in relation to (or "anchored to") the peak ground acceleration (PGA), which corresponds to a frequency of 100 hertz (Hz). A common measure of seismic fragility of an SSC is its median fragility value. The higher the median seismic fragility value of an SSC, the lower the failure probability of that SSC under seismic demand.

The SSCs in the realistic pathway include the MSL piping, the bypass and drain piping, and the main condenser. Several of these SSCs are nonsafety related. As noted in the 1999 SE, requiring the nonseismically analyzed portions of the main steam system piping and components to meet seismic Category I requirements would be impractical because the modifications required to upgrade the system to those requirements would be very costly. In addition, the guidance in RG 1.183 allows credit for the condenser, which is a nonsafety-related SSC, without any additional information or analysis from the licensee related to the "seismic robustness" at a plant's SSE for the deterministic dose analysis for the rod drop accident. However, the same RG does not provide credit for the condenser without further analysis for "seismic robustness" at a plant's SSE for the deterministic dose analysis for the MHA. The reason for the differing treatment of the same SSC under the same seismic loading (i.e., the plant's SSE) is unclear.

Multiple and diverse sources (References 15 through 18), including recently developed seismic probabilistic risk assessments (SPRAs; examples in References 19 through 25), have demonstrated that welded and bolted piping, such as MSLs and bypass and drain piping, have high median fragility values.[4] The sources use or compile the results of analytical methods (e.g., conservative deterministic failure margin and separation of variables) and consider earthquake experience for the fragility determination of various SSCs. Consideration of failure modes is inherent in the fragility determination process because the fragility of an SSC is dependent on the failure modes that a fragility analyst and plant systems analyst, in conjunction, consider to be limiting to the functionality of the SSC.

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[4] The NRC staff has not endorsed EPRI Report 30020000709. Citing this report as a source of information for fragility data does not constitute an endorsement of the report.

These sources document the high median seismic fragility of welded and bolted piping ranging from 1g to greater than 5g (anchored to PGA), with most of the data clustering around 2g. As examples, NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," issued August 1985 (Reference 15), provides median seismic fragilities of 2.5g for main steam piping, 2.2g for balance-of-plant piping, and 1.6g for reactor coolant system piping. The median fragility of motor-operated valves, considering various failure modes including failure of the yoke, is also documented to be high, with most of the data clustering around 2.5g. The median fragility for pipe hangers is reported as 1.46g in NUREG/CR-4550, Volume 4, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," issued December 1990 (Reference 16).

Due to the high probability of occurrence of loss of offsite power during seismic events, SPRAs do not model the main condenser. Therefore, documentation of the fragility of the main condenser is uncommon. However, the main condenser is a large "box" that, based on earthquake experience, is expected to have high seismic capacity. The main condenser is usually a seismic Category II structure, which would necessitate its anchorage being designed to avoid failure at the plant's design-basis seismic loads. In addition, the very large and heavy main condenser is anchored directly to the floor of the turbine building. The location, size, and weight of the main condenser adds to its capacity to withstand the seismic acceleration, especially at a plant's SSE. The readily available information about seismic fragility relevant to the main condenser is for the expansion joint for the circulating water piping connection to the condenser from Electric Power Research Institute (EPRI) Report 30020000709, "Seismic Probabilistic Risk Assessment Implementation Guide," issued December 2013 (Reference 17), with a median seismic fragility of 0.4g (with randomness variability [β_r] of 0.22 and epistemic uncertainty [β_u] of 0.22).

For the purposes of this assessment, the 0.4g median seismic fragility, with the β_r and β_u of 0.22, are used to determine failure probability at a plant's SSE. The intent of using these values is to use median fragility parameters that include the weakest link in the realistic pathway. The selected fragility parameters encompass various SSCs in the realistic pathway as well as their respective failure modes. The deterministic dose calculations assume a prescribed release amount of fission products from the condenser (i.e., the condenser is assumed to be "open"). Therefore, the use of the fragility parameters for the expansion joint represents a conservatism as compared to the seismic capacity of the remaining SSCs (such as piping and valves) in the realistic transport pathway.

The selected median fragility values would also address failure modes resulting from the collapse of the turbine building because the median fragility for turbine buildings (assuming nonsafety-related building construction) in the available information has a lower bound of 0.5g. The selected median fragility values for this assessment result in a high confidence of low probability of failure (95 percent confidence that failure probability is 5 percent or less) of approximately 0.2g. For context, the review-level earthquake for every nuclear power plant in the United States was at least 0.3g during the Individual Plant Examination of External Events effort. Further, the lowest median fragility that is repeatedly documented (in the cited source documents as well as recent SPRAs) is 0.3g (for ceramic insulators on offsite power lines). It is also worth noting that inclusion of the failure of the expansion joints represents a broader range of failure modes than previously considered for the realistic pathway.

SSEs for the majority of plants, especially BWRs, fall within 0.12g and 0.25g PGA. Using the selected median fragility parameters results in a failure probability ranging from 0.08 percent to

5 percent at and below the range of SSEs.[5] Therefore, even under the selected fragility parameters, the failure probability of SSCs in the realistic pathway at a plant's SSE would be low.

Post-earthquake walkdowns of nuclear power plants have also demonstrated the high seismic capacity of balance-of-plant components. Examples include the walkdowns performed for the nuclear power plants at Kashiwazaki-Kariwa in Japan and North Anna Power Station in the United States. Both the plants experienced beyond-design-basis earthquakes. EPRI documents its independent walkdown of Kashiwazaki-Kariwa in EPRI Report 1016317, "EPRI Independent Peer Review of the TEPCO Seismic Walkdown and Evaluation of the Kashiwazaki-Kariwa Nuclear Power Plants," issued January 2008 (Reference 23).[6] The results from the independent walkdown do not identify damage in the turbine building or piping connected to reinforced concrete, including snubbers and pipe hangers.

Shortly following the 2011 earthquake in Mineral, VA, both the Unit 1 and Unit 2 reactors at North Anna tripped, and the station experienced a loss of offsite power. Subsequent analysis indicated that the spectral and peak ground accelerations for the operating basis and design-basis earthquakes were exceeded at certain frequencies for a short period of time (3 seconds). The technical evaluation by the NRC Office of Nuclear Reactor Regulation related to the restart of North Anna after the occurrence of the earthquake (Reference 24) documents the licensee's walkdowns and the NRC staff's review of SSCs to determine damage and loss of functionality.

The evaluation states that the licensee performed inspections of piping and pipe supports, including checking for snubber damage, leakage of hydraulic fluid and bent piston rods, damage at rigid supports to identify deformation of support structure, deformation of pipe due to impact to support structure, damage of expansion joints, damage or leakage of piping and branch lines, and damage to pipe at building joints and interfaces between buildings. The licensee visually inspected welds, flanges, attachment lugs, and couplings. The NRC staff's review agreed with the licensee's basis for concluding that piping and pipe supports had not been damaged. The licensee also walked down and inspected safety-related balance-of-plant SSCs and did not find any loss of functionality, and the NRC agreed with this conclusion.

The Great Tohoku Earthquake of 2011 produced the highest recorded ground motions experienced by operating nuclear power reactors. The Onagawa site located to the northeast of Sendai, Japan, was the site closest to the earthquake epicenter and experienced PGAs exceeding 0.5g. These accelerations exceeded the facility design basis at certain frequencies. Unit 1, a GE BWR 4 design plant constructed by Toshiba, and Unit 3, a GE BWR 5 constructed by Toshiba and Hitachi, were operating at full power at the time of the earthquake. As documented in an International Atomic Energy Agency (IAEA) assessment report, "IAEA Mission to Onagawa Nuclear Power Station to Examine the Performance of Systems, Structures and Components following the Great East Japanese Earthquake and Tsunami," issued 2012 (Reference 25), the plants safely shut down without incident following the earthquake. Little damage was noted in the turbine building affecting the PCS. The IAEA team identified damage to the main turbine bearing bolts (due to stretching) and to the ends of the low-pressure turbine blades due to wear from relative motion between the rotor and casing. No

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5 The outcome at the fundamental frequency of various SSCs in the alternate pathway would be similar due to the use of the "spectral ratios" to scale the fragility from PGA to the frequency of interest.

6 The NRC has not endorsed EPRI Report 1016317. Citing this report as a source of information for insights from post-earthquake walkdown does not constitute an endorsement of the report.

damage to the steam piping was noted. Section 7.4 of the IAEA report states, "[t]he systems supporting the balance of plant did not suffer damage including the turbine bypass and turbine stop valves since they operated after the earthquake."

It is recognized that site characteristics, location of SSCs, and operational practices are important factors in the plant response to an earthquake. Therefore, this assessment uses the information from walkdowns of nuclear power plants presented in the preceding paragraphs to provide insights on the seismic capability of SSCs in the realistic pathway rather than definitive conclusions about potential earthquake impacts. The insights from these walkdowns reveal the appreciable seismic capacity of SSCs in nuclear power plants and the ability of both safety and nonsafety-related SSCs to remain functional during and after an SSE. Every operating nuclear power plant in the United States has performed a walkdown focused on identifying weaknesses in SSCs when exposed to seismic events (including beyond-design-basis seismic events), and several plants have performed an Expedited Seismic Evaluation Process (ESEP) review as part of post-Fukushima actions resulting from Near-Term Task Force (NTTF) Recommendation 2.3. The ESEP reviews were performed to demonstrate seismic margin and expedite plant safety enhancements through evaluations and potential near-term modifications of certain core and containment cooling equipment while more comprehensive plant seismic risk evaluations are being performed.

The staff notes that material degradation due to aging can result in reduction in seismic capacity of SSCs. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," issued December 2010 (Reference 19), provides the NRC staff's generic evaluation of the existing plant programs and documents the technical basis for determining existing programs that are adequate without modification and existing programs that should be augmented for the period of extended operation. The programs, with or without modification, are termed aging management programs (AMPs). Section VIII of NUREG-1801 discusses AMPs for the steam and power conversion system, including separate discussions for the main steam system (BWR), extraction steam system, condensate system, external surfaces of components and miscellaneous bolting, and common miscellaneous material/environment combinations. Section III.B2 of NUREG-1801 discusses supports for conduits and non-ASME piping and components, including anchorage and supports, and corresponding AMPs. Similarly, Section III.B1 discusses AMPs for supports for ASME piping and components. Therefore, material degradation due to aging in SSCs relevant to this assessment is appropriately addressed for licensees.

In summary, based on the available information and using the fragility parameters that represent various SSCs in the realistic path and their failure modes, the probability of the unavailability of the realistic pathway at a plant's SSE is low.

2.4 Occurrence Frequencies of Design-basis Seismic Events

The median fragility evaluation discussed in the previous section provides information about the failure probability of SSCs in the realistic pathway if an SSE were to occur. Using the plant-specific seismic hazard in conjunction with the median fragility parameters provides an indication of the frequency of occurrence of a radioactive release. Such an occurrence frequency can be determined by convolving the seismic hazard with the selected median fragility parameters. Such an approach assumes that every earthquake, even one at or below a plant's SSE, results in core damage.

Every operating nuclear power plant in the United States has performed a reevaluation of the plant-specific seismic hazard using present day information as part of post-Fukushima actions resulting from NTTF recommendations. Therefore, generic or assumed hazard curves are unnecessary. Since the median fragility parameters are "anchored to" the PGA, the hazard curve of interest would be the mean PGA hazard curve (i.e., the mean hazard curve for 100 Hz frequency).

It would be onerous and beyond the scope of this assessment to perform the convolution discussed above for every operating BWR. For the purposes of this assessment, the convolution was carried out for three BWRs with SSEs corresponding to 0.13g, 0.15g, and 0.24g (PGA). In each case, the convolution of the hazard and the selected median fragility parameters resulted in a cumulative occurrence frequency of failure of the SSCs in the realistic pathway on the order of magnitude of 1×10^{-6} considering even the entire hazard curve (i.e., beyond-design-basis earthquakes). The contribution from earthquakes at and below the SSE was less than 1×10^{-6} per year.[7] Therefore, even under the selected median fragility parameters and assumptions on accident initiation and progression, the risk of unavailability of the realistic pathway at a plant's SSE is low. Even under the assumption that failure of the realistic pathway results in the releases going directly to the control room or the environment, the occurrence frequency of radiological releases to the control room or to the public is low.

2.5 Uncertainty Evaluation

This assessment accounts for parametric uncertainty through the lognormal uncertainty factors for the selected median fragility, as discussed in Section 2.3 of this assessment. These uncertainty factors impact the estimates of the risk of unavailability of the realistic pathway discussed in Section 2.4 of this assessment. The fragility values in the sources referenced in Section 2.3 of this assessment are based on state-of-practice methods, which addresses modeling uncertainty in the median fragility values from the sources. As demonstrated in the previous sections, this assessment includes several conservatisms, such as the use of the selected median fragility and consideration of an SSE in conjunction with the MHA. These conservatisms further address modeling uncertainties related to the median fragility of various SSCs for seismically induced failure modes. In addition, conservatisms exist in the guidance for the deterministic dose calculation approach which are unchanged by this assessment.

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7 The results continue to remain valid using the so-called "simple average approach" from the efforts related to GI-199, as documented in "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (Reference 20).

APPENDIX B

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APPENDIX C

Analysis of Public Comments on Draft Interim Staff Guidance, "Supplemental Guidance for Radiological Consequence Analyses Using Alternative Source Terms"

Comments on the subject draft Interim Staff Guidance are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. The following table lists the comments the NRC received on the draft Interim Staff Guidance (ISG).

Comment Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
NRC-2021-0106-DRAFT-0003-1 through 13	ML21173A137	Nuclear Energy Institute	Frances A. Pimentel
NRC-2021-0106-DRAFT-0004	ML21173A138	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0005	ML21173A139	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0006	ML21173A140	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0007	ML21173A141	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0008	ML21173A142	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0009	ML21173A143	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0010	ML21173A145	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0011	ML21173A146	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0012	ML21173A149	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0013	ML21173A150	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0014	ML21173A153	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0015	ML21173A154	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0016	ML21173A155	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0017	ML21173A156	Anonymous	Anonymous

Comment Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
NRC-2021-0106-DRAFT-0018	ML21173A157	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0019	ML21173A158	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0020	ML21173A159	United States	Liberty Toussaint
NRC-2021-0106-DRAFT-0021	ML21173A160	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0022	ML21173A161	Anonymous	Anonymous
NRC-2021-0106-DRAFT-0023	ML21154A162	Anonymous	Anonymous

The following tables list each public comment by letter number, as given in the table above. The original comment as written by the commenter is provided. The comments are arranged in tables to group similar comments together. The first row of the table provides the general subject for the grouping.

Comments associated with purpose/applicability/background of ISG		
Comment Identifier	**Specific Comment**	**Staff Response**
NRC-2021-0106-DRAFT-0003-1	The text states: this ISG will help to resolve differences between the licensee's methods and assumptions and those deemed acceptable to the NRC staff when reviewing license amendment requests (LARs) that propose an increase in the MSIV leakage allowed by technical specifications (TS) for boiling water reactors (BWRs). Why is this ISG not applicable to new plant designs that meet current regulations and guidance? New plants have in many	The NRC staff disagrees with this comment. In new reactor applications, it is expected that an applicant would propose and justify a particular main steam isolation valve (MSIV) leakage rate value, which would be reviewed by the NRC staff. However, a new reactor licensee can propose, through a license amendment request (LAR), an increased MSIV leakage rate after receipt of its license. This ISG can be used by the NRC staff in its review of such a LAR. As stated in the "Applicability" section of the ISG, this ISG is applicable to 10 CFR Part 52 licensees. However, it is not applicable to 10 CFR Part 52 applicants due to the reasons identified above. No changes were made to the final ISG as a result of this comment.

	cases more robust designs and should be allowed a similar approach. Recommendation: Add words for anyone applying for a license such as for new reactor applications.	
NRC-2021-0106-DRAFT-0003-2	When is SRP 15.0.1 expected to be updated and will it apply to new plant designs?	It is expected that this guidance will be transitioned into the NRC standard review plan (SRP) Section 15.0.1 by September 2022. This transition is anticipated to occur in conjunction with a separate ongoing effort to revise Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” (ADAMS Accession No. ML003716792) which is expected to be complete by July 2022. Following the transition of this guidance into the SRP, this ISG will be closed. The NRC response regarding new plant designs is provided in comment NRC-2021-0106-DRAFT-0003-1. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0003-6	This section of the ISG provides why the ISG was developed and that thetechnical assessment identified the PCS as a realistic and available hold- up volume for fission products. The ISG also concludes that there is high confidence in the SSCs in the PCS to provide a volume for holdup and retention of fission products. The question is, why isn’t the Staff	This comment is out of scope of the ISG because it relates to RG 1.183, which is a separate document. Once the draft of RG 1.183 is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.

	incorporating this well thought out, risk-informed methodology into Regulatory Guide (RG) 1.183. Why is this method being limited for use only by the NRC Staff when reviewing AST LARs?	
NRC-2021-0106-DRAFT-0003-8	Does the ISG apply to RG 1.183 Rev. 0 or Rev. 1 or both?	The approach in this ISG will apply to both RG 1.183, Rev. 0 and Rev. 1 if the licensee's dose analysis does not include quantitative credit for the so-called alternate pathway using the guidance in those RGs. The technical assessment in the ISG is independent of the revision of RG 1.183 being used. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0003-9	Can the ISG be applied to new reactors? New reactors may have additional pathways available.	The NRC response regarding applicability of the ISG to new reactors is provided in response to comment NRC-2021-0106-DRAFT-0003-1. The ISG does not provide any accepted methods to demonstrate a new reactor facility would comply with regulatory requirements. The term alternate or alternative pathway has a specific meaning in the context of RG 1.183. The ISG is not restricted to that pathway. No changes were made to the final ISG as a result of this comment.
Comments associated with formal credit to applicant's design basis		
NRC-2021-0106-DRAFT-0003-4	There have been numerous seismic industry initiatives that demonstrated that SSCs have sufficient seismic capacities to withstand accelerations associated with the SSE and higher. All Exelon sites were evaluated to more	This comment is out of scope of the ISG as it relates to RG 1.183, which is currently being updated by the NRC staff. Once the draft of RG 1.183 is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.

	modern seismic information as part of NTTF Recommendations 2.1 and 2.3 from Fukushima Dai-Ichi accident which included plant walkdowns, expedited seismic evaluation programs (ESEPs) and Seismic Probabilistic Risk Assessments (SPRAs). These evaluations used the Seismic Hazard Curves (SHC) developed by EPRI in 2013. Some plants did a SPRA, some did a Seismic Margin Assessment as documented in EPRI NP-6041-SL "A Methodology for Assessment of Nuclear Power Plant Seismic Margin." Also, Individual Plant Examination of External Events (IPEEE) and Unresolved Safety Issue (USI) A-46, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors," results demonstrated that safe shutdown Structures, Systems, and Components (SSCs) are capable of withstanding accelerations in excess of SSE	

	loading. All of these assessed a lot of Class 1 piping and structures, and selected safety related seismically qualified devices. If those risk-based approaches are used to credit the robustness of the MSIV leakage alternate pathway SSCs, will the NRC accept these assessments or will they request implementation of a backfit for a more updated PRA quality assessment (i.e., RG 1.200, ASME/ANS Joint Standard) be performed, including updated SHCs or Ground Motion Response Spectra (GMRS)?	
NRC-2021-0106-DRAFT-0003-5	This section discusses four approved LARS from 2019 that increased MSIV leakage. What If other plants in the future want to do the same -what would industry need to submit to ensure future applications can get credit for the risk insight included in the ISG?	The ISG provides support for the NRC staff to reach a reasonable assurance conclusion for approving increased MSIV leakage when evaluating traditional deterministic analyses in support of a license amendment containing parameters with associated uncertainty. The technical assessment for the ISG does not rely on plant-specific information. Therefore, the ISG does not require licensees to submit any specific information for the NRC staff to be able to use the risk insights contained in this ISG. The licensees for the four approved LARs referred to in the ISG provided sensitivity analyses; however, these analyses were not requested by the staff to use the information in the ISG. No changes were made to the final ISG as a result of this comment.

NRC-2021-0106-DRAFT-0003-7	How can we get credit for other risk-insights mentioned in this ISG? Why are these not being considered for incorporation in the future revision of RG 1.183.	This comment is out of scope of the ISG because it relates to RG 1.183, which is a separate document. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0003-12	Is EPRI 3002017583 and 3002012988 “Alternative Approaches for Addressing Seismic risk in 10CFR50.69 Risk-Informed Categorization,” July 2018 an acceptable option for justifying credit for the alternate pathway? This graded (alternate) approach developed by EPRI is based on comparison of the site SSE and the latest GMRS from Seismic Hazard Analysis.	Credit for the alternate pathway in dose calculations is addressed in RG 1.183, which is a separate document from the ISG. Therefore, this comment is out of scope of this ISG. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0003-13	Appendix A points out that RG 1.183 itself is inconsistent in that it allows credit for the condenser, which is a non-safety-related SSC, without any additional information or analysis from the licensee related to the “seismic robustness” at a plant’s SSE for the	This comment is out of scope of the ISG because it relates to RG 1.183, which is a separate document. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.

	deterministic dose analysis for the rod drop accident. However, the same RG does not credit the condenser without further analysis for “seismic robustness” at a plant’s SSE for the deterministic dose analysis for the maximum hypothetical accident (MHA). Since there is no reason for the different treatment, why doesn’t the Staff revise the RG to allow credit for the condenser for the MHA based on the risk-informed conclusions provided by the ISG technical assessment?	
Comment out of scope regarding use of risk insights		
NRC-2021-0106-DRAFT-0003-3	The licensing basis radiological analyses have typically been performed by deterministic evaluations with all inputs biased in the conservative direction for a very bounding result. Other safety analyses, such as the GSI-191 resolution or 50.46 LOCA analysis, have applied risk-informed or statistical approaches in order to provide more-realistic but	The ISG does not change or risk-inform the licensee’s radiological dose analyses. It provides support for the NRC staff to reach a reasonable assurance conclusion when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Expanded use of risk insights for licensing basis radiological dose analyses falls in the purview of RG 1.183, because it deals with potential changes to the acceptable methods for performing the dose analyses. Therefore, this comment is out of the scope of this ISG. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.

	bounding results while maintaining defense-in-depth. The application of risk- informed concepts from the ISG is a good start to make the MHA dose analysis more consistent with other safety analyses. In what other areas is the NRC considering risk-informing the licensing basis radiological dose analyses applying the Staff's LIC-206 process for multi-disciplinary risk-insights?	
NRC-2021-0106-DRAFT-0003-11	In this section, the ISG references PRAB-02-O1, Assessment of BWR Main Steam Line Release Consequences, John N. Ridgely, October 2002. This document indicates that MSIV leakage rate orders of magnitude higher than what is currently allowed would be necessary to result in exceeding the dose limits associated with the Commission's Safety Goal Policy Statement (51 FR 30028 (August 21, 1986)). Specifically, in the conclusion it states, "The second objective was to determine the	The ISG does not specify or change the acceptable methods for performing dose calculations for MSIV leakage and therefore, the value of the proposed increase in LARs. It provides support for the NRC staff to reach a reasonable assurance conclusion when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Acceptable methods for calculations for MSIV leakage and therefore, the value of the proposed increase in LARs falls in the purview of RG 1.183, which is a separate document. Therefore, this comment is out of the scope of this ISG. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. No changes were made to the final ISG as a result of this comment.

	leakage rate which, if achieved in a plant, should result in some additional investigation by the NRC and the licensee as to the potential consequences of a postulated accident concurrent with the increased leakage. […] Given the factor of conservatism being greater than 20, the conclusion is made that there should be no regulatory concern if the leakage past the best sealing valve in the main steam line is less than 10,000 scfh." Based on this assessments conclusion, why doesn't Staff approve and allow the industry to utilize higher MSIV leakage values in our calculations associated with source term?	
NRC-2021-0106-DRAFT-0011	The NRC is confusing the use of PRA and quantitative health objectives with the intent of the regulations in 10 CFR Part 20 and 10 CFR 50.67. Limits on radiation dose for protecting workers and the public are based on preventing the likelihood of cancer in the	The NRC staff disagrees with the comment. The ISG cannot and does not change any regulations. It does not specify or change the acceptable methods for performing dose calculations to meet the regulations in 10 CFR Part 20 and at 10 CFR 50.67. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Further, the technical assessment for the ISG uses neither a probabilistic risk

	exposed population. The proposed ISG would adversely affect the dose to workers, and especially control room operators charged with mitigating the consequence of accidents, and have adverse impacts on public health and safety and protecting the environment.	assessment as defined in NUREG-2122 nor does it rely on the quantitative health objectives. No changes were made to the final ISG as a result of this comment.
Comments regarding ISG and RG 1.183 review process		
NRC-2021-0106-DRAFT-0004	The draft ISG represents a substantial departure from the design basis of boiling water reactors and should be reviewed by the Atomic Safety Licensing Board Panel.	The staff disagrees with the comment. The ISG does not change or result in departures from the design basis of boiling water reactors. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67 or any system qualifications. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. NRC's process for interim staff guidance documents is being followed for this ISG. This process, described in LIC-508, "Development, Issuance, and Closure of Interim Staff Guidance Documents for the Office of Nuclear Reactor Regulation," involves internal review, review by the Advisory Committee for Reactor Safeguards (ACRS), the Office of General Counsel (OGC), and members of the public. The jurisdiction of the Atomic Safety and Licensing Board Panel (ASLBP) is generally limited to reviewing contested matters in NRC adjudications. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0010	It is improper to change the accident analysis using NRC	The staff disagrees with this comment.

	review guidance such as the proposed ISG. Chapter 15 accident analysis are conservative and are meant to be bounding analysis. Changing the basic premises of the accident analysis using PRA is a technical decision that should be reviewed by the ACRS, a legal decision that should be reviewed by the ASLB, and a policy decision that should be reviewed by the Commission. The NRC should submit a Notation Vote paper to the Commission on this matter.	The ISG cannot and does not change any regulations. The ISG also does not change the accident analysis. It does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the requirement that the licensee's analysis of record demonstrates with reasonable assurance that the acceptance criteria in 10 CFR 50.67 will not be exceeded. Rather, it follows existing Commission policy direction on using risk insights in staff reviews and is consistent with previous staff reviews in this area. Therefore, it provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. NRC's process for interim staff guidance documents, per NRR's office instruction, LIC-508, is being followed for this ISG, which includes review by the ACRS and OGC. OGC reviewed the draft ISG prior to issuance for public comments. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0019	The ISG concurrence appears to only have had technical and legal reviews. Has the NRR licensing organization agreed to the proposed ISG? Is there an NRC-approve topical report under review to support the positions in the ISG? Has RG 1.183 been revised to support the new NRC positions in the ISG? Is there a policy paper to the Commission requesting to approval of this	The ISG cannot and does not change any regulations. It does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67. Rather, it follows existing Commission policy direction on using risk insights in staff reviews and is consistent with previous staff reviews in this area. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. As a result, (1) the NRC staff does not need to approve a topical report to utilize the ISG for its own decision making, and (2) no policy papers need to be presented to the Commission for this ISG. The question about changes to RG 1.183 is out of scope of this ISG because the RG is a

	change in policy/approach concerning the plant design basis?	separate document. RG 1.183 is currently being updated by the NRC staff. Once the draft of RG 1.183 revision is complete, the public will be given an opportunity to provide comment on the guidance document prior to issuance. NRC's process for interim staff guidance documents, per NRR's office instruction, LIC-508, is being followed for this ISG, which involves internal review, review by the ACRS, OGC, and review by members of the public. No changes were made to the final ISG as a result of this comment.
Comment Question		
NRC-2021-0106-DRAFT-0003-10	The end of the first paragraph discusses that this ISG is a structured evaluation of the acceptability of dose consequence analyses for MSIV leakage when the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameter used in the deterministic dose calculations. Which input parameters had challenges due to uncertainty? Please provide more detail.	Dose calculations include several parameters which have associated uncertainty. During LAR reviews, the staff encountered challenges related to uncertainties in the modeling of drywell spray aerosol removal and aerosol deposition. The ISG is not focused on the uncertainty associated with a particular parameter. The technical assessment and the purpose of the ISG is broad enough such that it can be used by NRC staff to address challenges associated with uncertainty in the dose calculation parameters. No changes were made to the final ISG as a result of this comment.
Comments implying or stating that the ISG should not be approved		
NRC-2021-0106-DRAFT-0005	The propose ISG allows plant to operate with	The staff disagrees with this comment.

	degraded safety equipment credited for mitigating the release of radiation adversely affecting operators in the control room. The ISG credit for condenser holdup is non-conservative and is not supported by experimental testing for the removal of radionuclides. As such, the ISG would not meet the regulations in 10 CFR 50.67 for control room dose. The ISG should not be approved. ~~The draft ISG represents a substantial departure from the design basis of boiling water reactors and should be reviewed by the Atomic Safety Licensing Board Panel.~~	The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67. This ISG provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. The ISG does not provide any quantitative "credit" for condenser holdup, which would change the licensee's analysis of record. Quantitative "credit" for condenser holdup is the purview of RG 1.183. ~~Regarding the departure from design basis and review by the ASLB, please refer to comment response NRC-2021-0106-DRAFT-0004.~~ No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0006	The ISG advocates an improper use of risk analysis that represents an industry bias and loss of objectivity. The NRC should not misuse risk analysis to advocate for the industry a justification for not fixing MSIVs that leak.	The staff disagrees with this comment. This ISG follows existing Commission policy direction on using risk insights and is consistent with previous staff reviews in this area. Further, the ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67. The licensing basis for operating plants allows a certain amount of MSIV leakage via provisions in the plant-specific technical specifications. Licensees have the ability to request changes to the licensing basis, including the amount of MSIV leakage per technical specifications, via an LAR submitted under 10 CFR 50.90.

		No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0007	The NRC should not change the design basis of all operating BWRs using review guidance. The use of any review guidance for changes to the current licensing basis should follow NRC-approved methods (e.g., Regulatory Guide 1.183) not precede the approval of methods. The current ISG does not meet the current version of RG 1.183 and should not be approved.	The staff disagrees with this comment. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG neither approves any "methods" for use in the dose calculations nor does it alter the licensee's analysis of record to meet 10 CFR 50.67. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Therefore, the ISG neither conflicts with RG 1.183, Revision 0, nor does it rely on the issuance of RG 1.183, Revision 1. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0018	The proposed ISG appears to be an aggressive attack on the regulations intended to protect workers and the public from exposure to radiation. The use of PRA to compensate for licensee's inability to maintain the plant, and barriers to the release of radiation (i.e., MSIVs), seems exceedingly inappropriate, improper, and possibly illegal. Enforcement for failure to maintain MSIVs would seem more appropriate. ~~The propose ISG allows plant to operate with~~	The staff disagrees with this comment. The licensing basis for operating plants allows a certain amount of MSIV leakage per provisions in the plant-specific technical specifications. Licensees have the ability to request changes to the licensing basis, including the amount of MSIV leakage per technical specifications, using a LAR submitted under 10 CFR 50.90. A licensee that fails to meet its technical specifications would be subject to enforcement action. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the licensee's analysis of record to meet 10 CFR 50.67. The ISG does not provide any quantitative "credit" for condenser holdup, which would change the licensee's analysis of record. Quantitative "credit" for condenser holdup is the purview of RG 1.183. Rather, the ISG supports the staff's use of risk insights in resolving uncertainties associated with deterministic analyses supporting license amendment

	~~degraded safety equipment credited for mitigating the release of radiation adversely affecting operators in the control room. The ISG credit for condenser holdup is non-conservative and is not supported by experimental testing for the removal of radionuclides. As such, the ISG would not meet the regulations in 10 CFR 50.67 for control room dose. The ISG should not be approved.~~	requests to increase allowed MSIV leakage rates. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0012	The proposed SG would not meet the NRC-approved methods in Regulatory Guide (RG) 1.183 for alternate source terms. The ISG should not be implemented until after a revision of RG 1.183 is approved for use by the industry.	The staff disagrees with this comment. The ISG will be used by the NRC staff to support their reasonable assurance findings, whereas the RG provides one acceptable method to meet regulations. The ISG does not specify or change the acceptable methods in RG 1.183 for performing dose calculations to meet the regulations. The ISG does not alter any licensee's analysis of record to meet 10 CFR 50.67. The ISG is independent of the revision to RG 1.183. Further, process for interim staff guidance documents, per NRR's office instruction, LIC-508, is being followed for this ISG. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0013	The proposed ISG is risk-based, not risk-informed, and should not be approved.	The staff disagrees with this comment. The ISG uses engineering and risk insights that the staff can use in conjunction with the licensee's analysis of record to make a reasonable assurance determination. Refer to the response to NRC-2021-0106-DRAFT-008. No changes were made to the final ISG as a result of this comment.

NRC-2021-0106-DRAFT-0016	The ISG lacks citations and references to realistic experimental testing to support the decontamination factors and holdup of accident fission products. The radiological analysis is contrived. The ISG should not be issued until radiological test data is subject to a public discussion/debate and review by the ACRS.	The staff disagrees with this comment. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not provide any quantitative "credit" for condenser holdup, which would change the licensee's analysis of record. Therefore, the ISG does not provide any decontamination factors that can be used by the licensee. Quantitative "credit" and resulting decontamination factors for condenser holdup is in the purview of RG 1.183 and out of scope of this ISG. Refer to the response to comment NRC-2021-0106-DRAFT-0010 for a response on ACRS review. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0021	The ISG relies on undemonstrated and speculative decontamination factors and holdup assumptions, and is, therefore, unacceptable. The ISG should not be approved. ~~The proposed SG would not meet the NRC-approved methods in Regulatory Guide (RG) 1.183 for alternate source terms. The ISG should not be implemented until after a revision of RG 1.183 is approved for use by the industry.~~	The staff disagrees with this comment. Refer to responses to ~~NRC-2021-0106-DRAFT-0012 and~~ NRC-2021-0106-DRAFT-0016. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0022	The ISG improperly accounts for release points and the likelihood that control room dose will be exceeded. The ISG also fails to consider the poor	The staff disagrees with this comment. The considerations identified in the comment are either part of RG 1.183 to perform dose calculations or out of scope of those calculations. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The

	maintenance practices for control room emergency air filtration systems, pre-conditioning of emergency filtration systems to pass surveillance tests, and plant-specific challenges for control room habitability. ~~The proposed ISG is risk-based, not risk-informed, and should not be approved.~~	ISG neither approves any “methods” for use in the dose calculations nor does it alter the licensee’s analysis of record to meet 10 CFR 50.67. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Therefore, the considerations identified in the comment are either part of the RG to perform dose calculations or out of scope of this ISG. ~~Refer to the response to NRC-2021-0106-DRAFT-008 for the response to the comment on the ISG not being risk-informed.~~ No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0023	The NRC staff's approach in the ISG is inappropriately weighted by looking narrowly at the safe shutdown earthquake (SSE) at the exclusion of other initiating events. It is not apparent that control room dose would be met by the NRC staff's rationale in the ISG. The full spectrum of accidents and mitigating systems in Chapters 6 and 15 of the UFSAR should be analyzed. Presumed holdup characteristics may not be realistic, particularly if human error is considered. The ISG lacks citations and references to realistic experimental testing to support the	The staff disagrees with this comment. The regulations in 10 CFR Part 50 and Part 100 apply qualification for the safe shutdown earthquake. Part 50, Appendix B, applies quality assurance of the design, fabrication, construction, and testing as fundamental measures to assure the reliability of structures, systems, and components (SSCs). Accordingly, the staff considered these factors for the power conversion system in assessing the reliability of the main steam and connected systems for the limited function of maintaining a low-pressure boundary for holdup of releases that leak by the MSIVs. The staff found ample evidence that the main steam systems at BWRs had been subject to enhanced quality assurance measures and that the components were sufficiently robust to withstand seismic accelerations. Protection against other design basis natural phenomena and equipment failures (other than LOCAs) is limited to safe shutdown equipment and excludes accident mitigation equipment based on the low probability that such an event results in damage to a fission product barrier. Therefore, the staff concluded that there is high probability that the SSCs in the PCS will be available to mitigate releases due to MSIV leakage. Consequently, the ISG provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating

	decontamination factors and holdup of accident fission products. The radiological analysis is contrived. The ISG should not be issued until radiological test data is subject to a public discussion/debate and review by the ACRS.	traditional deterministic analyses containing parameters with associated uncertainty without altering the licensee's analysis of record. Refer to the response to NRC-2021-0106-DRAFT-0016 for the response on decontamination factors. Refer to response to NRC-2021-0106-DRAFT-0010 for response on review by the ACRS and the public, which is part of the formal ISG process that is being followed. No changes were made to the final ISG as a result of this comment.
Comments regarding use of RG 1.174		
NRC-2021-0106-DRAFT-0008	The proposed ISG does not meet the principles of Regulatory Guide 1.174 and 1.177 for risk-informed changes to the current licensing basis. In particular, defense-in-depth and safety margins are not maintained in allowing barriers to be release to release of radioactive material to degraded.	The staff does not agree with this comment. The ISG uses engineering and risk insights to provide support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Therefore, the ISG is not the sole decision-making basis for the MSIV leakage analysis review. MSIV leakage LARs are not formally risk-informed using the guidance in RG 1.174 and 1.177. Further, the use of risk insights by the staff is not limited to risk-informed LARs, as demonstrated by LIC-206. Because the ISG is going to be used to support the staff's reasonable assurance findings when evaluating traditional deterministic MSIV leakage analyses containing parameters with associated uncertainty, it needs to be used in conjunction with the deterministic analysis reviews which provide primary basis. Defense-in-depth exists in the dose analysis because of the assumption of the maximum hypothetical accident (MHA). The postulated assumptions of failure of all Emergency Core Cooling Systems (ECCS) leading to the source term add to the defense-in-depth considered in the analysis (i.e., the probability of failure of independent and redundant ECCS and non-safety systems is not considered). Conservative assumptions used in the guidance for performing

		the dose analysis adds to the defense-in-depth. The ISG considers the failure probability of the balance-of-plant SSCs due to an SSE coincident with the MHA which adds to the defense-in-depth already built into the analysis. Safety margins continue to exist in the codes of record as well as accident analysis assumptions. Further, the ISG also includes margin by not taking any advantage of the fact that piping designed for high pressure will be subjected to seismic loading at low pressure conditions of the MHA. Performance monitoring is achieved via existing technical specifications and licensee programs for safety- and non-safety related SSCs. Therefore, even though MSIV leakage LARs do not follow RGs 1.174 and 1.177, risk insights can be used for these LAR reviews per NRR's office instruction, LIC-206, and the five principles of risk-informed decision making can be applied in these reviews. No change were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0015	The NRC is improperly citing the Staff Requirements Memoranda for SECY-98-144 and SECY-19-0036 in justifying relaxation of MSIV requirements. The proposed ISG does not meet the principles for risk-informed changes to the licensing basis in Regulatory Guide 1.174.	The staff disagrees with this comment. Licensees have the ability to request changes to the licensing basis, including the amount of MSIV leakage per technical specifications, using a LAR submitted under 10 CFR 50.90. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. The ISG does not alter the requirement that the licensee's analysis of record demonstrates with reasonable assurance that the acceptance criteria in 10 CFR 50.67 will not be exceeded. The ISG cannot and does not change the acceptance criteria in 10 CFR 50.67. Therefore, the ISG does not justify "relaxation of MSIV requirements." Rather, the ISG appropriately relies on Commission policy direction in SRM-SECY-98-144 and SRM-SECY-19-0036 for the staff to realistically consider risk insights in conjunction with its review of licensee analyses.

		See response to NRC-2021-0106-DRAFT-0008 on consistency with the principles of risk-informed decision making in RG 1.174. No changes were made to the final ISG as a result of this comment.
Comments that ISG is contrary to current regulations		
NRC-2021-0106-DRAFT-0009	The proposed ISG is contrary to the regulations in 10 CFR 50.2, Design Basis. The ISG undermines the integrity of functional goals derived from analysis and the effects of SSC performance to meet those functional goals.	The staff disagrees with this comment. The ISG does not undermine the integrity of functional goals derived from analysis and the effects of SSC performance to meet those functional goals. Required regulatory analyses must continue to be performed by the licensees. This ISG supports staff consideration of risk insights when reviewing such analyses. No changes were made to the final ISG as a result of this comment.
NRC-2021-0106-DRAFT-0014	The proposed ISG is supported by analysis prepared by the NRC staff and does not rely on information submitted by licensees to be incorporated in the plant licensing basis. As such, the ISG is proposing to incorporate information in the licensing basis contrary to 10 CFR 54.3, "Current Licensing Basis," that specifies "...information made in docketed licensing correspondence..." becomes part of the licensing basis. NRC inclusion of risk information, not submitted on the docket, would not become part of the	The staff does not agree with this comment. The ISG does not specify or change the acceptable methods for performing dose calculations to meet the regulations. This is explicitly stated in the "Rationale" section of the ISG. The ISG does not alter the requirement that the licensee's analysis of record demonstrates with reasonable assurance that the acceptance criteria in 10 CFR 50.67 will not be exceeded. It provides support for the NRC staff to reach a reasonable assurance conclusion that the applicable regulations have been met when evaluating traditional deterministic analyses containing parameters with associated uncertainty. Because the ISG does not change the analysis of record, the ISG does not propose to incorporate any information in the licensing basis that is not submitted by the licensee. No changes were made to the final ISG as a result of this comment.

	current licensing basis.	
NRC-2021-0106-DRAFT-0017	The NRC staff's approach to handling uncertainties in the ISG is deficient in that it only addresses parameter uncertainties. Modeling uncertainties are ignored, altogether, when it is generally known that modeling uncertainties dominate parameter uncertainties in the evaluation of risk.	While the staff does not agree with this comment, Section 2.5 of Appendix A of the ISG was updated for clarity. Section 2.5 of the Appendix A to the ISG explicitly addresses parametric and modeling uncertainties in the selected lower bound median fragility value. Parametric uncertainty is accounted for through the use of the lognormal uncertainty factors for the lower bound median fragility. Conservatisms are included within the ISG as well as the unchanged conservatisms in the deterministic dose calculations that address modeling uncertainties. Further, as noted in RG 1.174, modeling uncertainty arises when “the industry’s state of knowledge is incomplete, and opinions may vary on how the models should be formulated.” The fragility values in the sources referenced in Appendix A to the ISG were developed using state-of-practice methods, which also addresses modeling uncertainty. Section 2.5 of Appendix A of the ISG was updated for clarity based on the comment on uncertainties.
NRC-2021-0106-DRAFT-0020	See attached file(s)	The information provided by the commenter is related to a 2020 XIAOMI Corporation Interim Report (ADAMS Accession No. ML21173A159) which is not relevant to this ISG. No changes were made to the final ISG as a result of this comment.