DRA-ISG-2021-XX

Supplemental Guidance for Radiological Consequence Analyses Using Alternative Source Terms

Draft Interim Staff Guidance

May 2021

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DRAFT INTERIM STAFF GUIDANCE

Supplemental Guidance for Radiological Consequence Analyses Using Alternative Source Terms

DRA-ISG-2021-XX

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).

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).

Consistent with Commission direction on risk-informed and performance-based regulation (e.g., Staff Requirements Memorandum (SRM)-SECY-98-144 at ADAMS Accession No. ML003753601) and 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 considering feedback from stakeholders, the staff evaluated whether modern analysis approaches and operating experience gained since the approval of 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), could be used to inform the reviews of the MSIV leakage increase LARs.

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 regulation 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 the 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. In addition, in SRM-SECY-19-0036, dated July 2, 2019, 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.”

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 power reactor early site permit, 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

Discussion to be provided in the final ISG.

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, the 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

APPENDIX A

Technical Assessment Supporting the Interim Staff Guidance

This technical assessment provides the basis for DRA-ISG-2021-XX 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 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. A realistic consideration of the typical configuration of a BWR main steam system provides holdup volumes for fission product retention and decay. The NRC staff has previously approved alternative methods for showing compliance with the regulation for the MSIV leakage pathway. In 1999, the NRC staff approved credit for the so-called 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 assessment includes consideration of the likelihood and consequence, in the form of an undesirable outcome, of 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

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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.

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.

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 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.

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, conservatisms in this assessment provide additional defense in depth and maintain the safety margin.

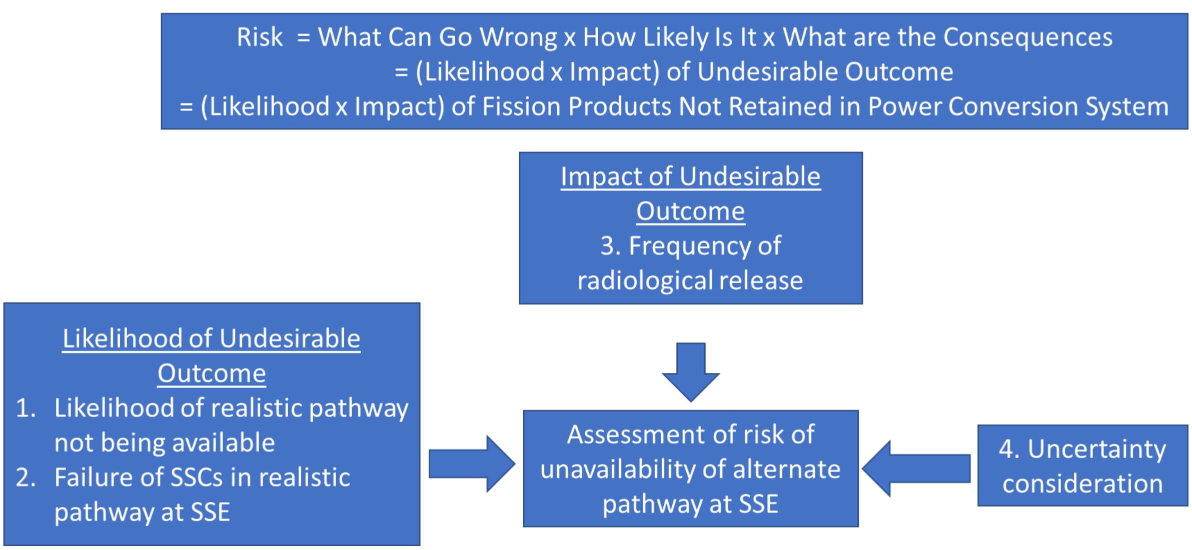

Figure 1. Assessment approach

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; Regulatory Issue Summary 2006-04, “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 generally does 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)[2] 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 realistic pathways for fission product leakage.

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[2] 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.

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 established, including relevant procedural changes, and that it would be "seismically robust" at the plant's SSE.

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.

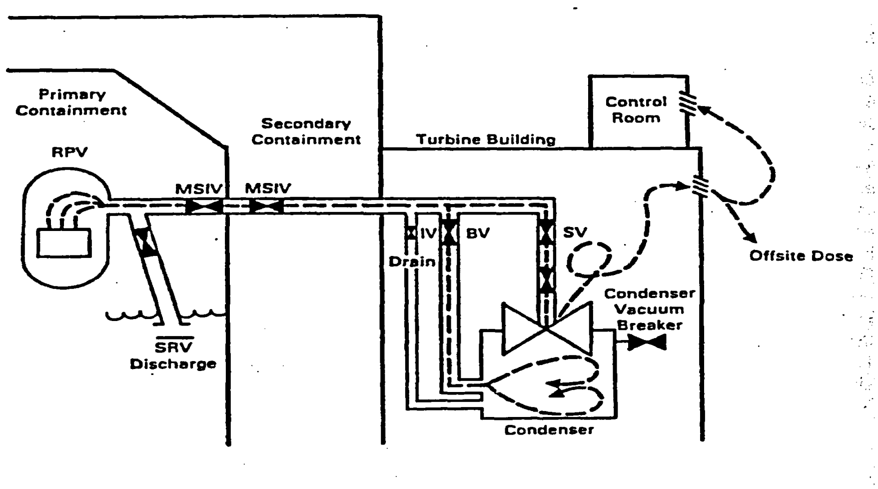

Figure 2. Schematic illustration of the alternate pathway (reproduced from Reference 11)

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.

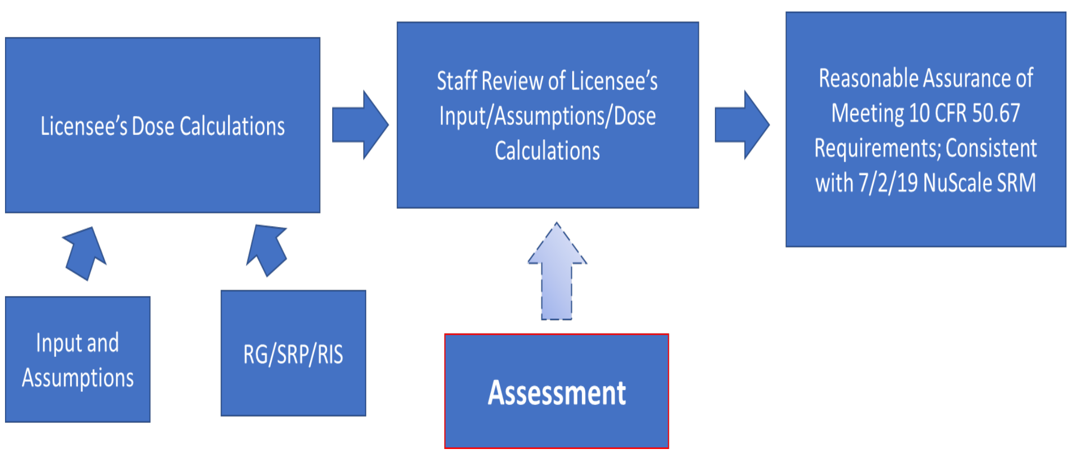

Figure 3. Relationship of this assessment to MSIV leakage dose consequence analyses

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 expenditure 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 9×10^{-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 1.2×10^{-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. Note that 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.[3] The sources either use or compile the results of analytical methods (e.g., conservative deterministic failure margin and separation of variables) and 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.

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,

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3 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.

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.[4] 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).[5] 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.

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4 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.

5 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.

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 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 addressed for licensees that currently have extended operating licenses or will apply for such licenses in the future.

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 (or a subset thereof). 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.[6] Therefore, even under the selected median fragility

[6] 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).

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

As demonstrated in the previous sections, the assessment summarized in this document includes several conservatisms, such as the use of the selected median fragility and consideration of an SSE in conjunction with the MHA. These conservatisms address uncertainties in the assessment by bounding the seismic failure probabilities of various SSCs and corresponding seismically induced failure modes. It is worth noting that the calculation of the failure probability using the median fragility parameters includes consideration of uncertainty in that parameter.

In addition, conservatisms exist in the postulated deterministic dose calculation approach. The Statements of Consideration accompanying the publication of 10 CFR 50.67 (Volume 99 of the *Federal* Register, page 33283) clarify that the design-basis accidents analyzed for dose calculations "are intentionally conservative in order to address uncertainties in accident progression, fission product transport, and atmospheric dispersion."

APPENDIX B

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