1 Introduction and Background

Reactor vessel internals (RVI) aging management programs are organized through industry-led initiatives, such as the Boiling Water Reactor Vessel Internals Program (BWRVIP) and the Materials Reliability Program (MRP). The Electric Power Research Institute (EPRI) manages these initiatives and publishes topical reports that describe and justify industry positions on inspection and evaluation of RVI. Some topical reports are reviewed and approved by NRC safety evaluation. EPRI maintains ongoing research activities to ensure continued effectiveness of the RVI aging management programs, as new information becomes available.

EPRI contacted NRC about new fracture toughness testing of irradiated stainless steel harvested from a decommissioned international reactor. At a May 27, 2021 public meeting, EPRI described new data that indicated the flaw evaluation procedures recommended in the BWRVIP guidance document, BWRVIP-100, Revision 1-A, may be nonconservative over a certain fluence range. BWRVIP-100, Revision 1-A describes recommendations for fracture toughness and evaluation of flaws in a BWR core shroud. The recommendations in BWRVIP-100, Revision 1-A for the shroud may also be applied in other related areas, such as other RVI components and MRP documents. The purpose of this document is to evaluate the risk significance of the nonconservatism and recommend agency actions to disposition the issue.

2 Applicable Regulations

For RVI, the following regulations are applicable:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC)
    - GDC 4, "Environmental and Dynamic Effects Design Bases"
    - GDC 27, "Combined Reactivity Control Systems Capability"
    - GDC 35, "Emergency Core Cooling" (considered together with 10 CFR 50.46)
- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"
- 10 CFR 50.55a, "Codes and Standards"
- 10 CFR 54.21, "Contents of Application – Technical Information"

10 CFR Part 50, Appendix A GDC 4 states the following:

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects

associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

In GDC 27, the NRC states that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

In 10 CFR 50.46(b)(4), the NRC states that calculated changes in core geometry shall be such that the core remains amenable to cooling. GDC 35 states, "A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."

American Society of Mechanical Engineers (ASME) Code, Section XI, which is incorporated by reference in 10 CFR 50.55a, requires visual VT-3 examinations of all RVIs classified as "core support structures" once each 10-year inspection interval.

The requirements in 10 CFR 54.21 specify that the aging of systems, structures, and components (SSCs) within the scope of license renewal are to be managed to maintain intended functions consistent with the current licensing basis for the period of extended operation.

3 Staff Evaluation

3.1 Issue Characterization

BWRVIP-100, Revision 1-A Guidelines

BWRVIP-100, Revision 1-A describes fracture toughness results for irradiated stainless-steel materials. Fracture toughness is an engineering parameter that describes the material resistance to crack propagation. Depending on a range of variables, cracks may propagate through materials in a ductile manner or a brittle manner. Engineering analysis procedures exist for predicting crack stability in engineering structures for both ductile and brittle behavior. BWRVIP-100, Revision 1-A provides recommendations for determining appropriate failure mode based upon neutron exposure to the material. Further, BWRVIP-100, Revision 1-A describes appropriate fracture toughness parameters to apply to BWR core shroud flaw evaluation. These recommendations allow licensees to justify inspection frequencies for RVI components.

The assumed toughness of the core shroud is based upon fracture toughness experiments described in BWRVIP-100, Revision 1-A. The results of these experiments illustrate how toughness and failure mode depend upon the fluence level. BWRVIP-100, Revision 1-A stated that brittle fracture was observed in test specimens above a certain neutron fluence threshold. For the test specimens that exhibited brittle fracture, the BWRVIP recorded the critical stress intensity factor, K_{Ic}. BWRVIP-100, Revision 1-A provides a recommended toughness for RVI flaw evaluation assuming brittle fracture. For test specimens that exhibited ductile tearing, the BWRVIP developed a bounding relationship for the toughness curve, known as J_R.

Based upon the test results and additional analyses, BWRVIP-100, Revision 1-A provides recommendations for flaw evaluation in BWR core shrouds. The document provides recommended fluence thresholds for brittle fracture and ductile fracture conditions. It further recommends that ASME Code, Section XI procedures, such as Nonmandatory Appendix K for elastic-plastic fracture mechanics, be applied when determining inspection frequency.

New Fracture Toughness Data

During a May 27, 2021 public meeting, EPRI described analysis of fracture mechanics testing results of irradiated stainless steel material from a decommissioned international reactor not considered in the development of the BWRVIP-100, Revision 1-A guidelines. The new data indicated that stainless steel weld metal may exhibit brittle fracture at a lower fluence threshold. Furthermore, EPRI suggested that the BWRVIP-100, Revision 1-A J_R curve may need to be adjusted to account for lower toughness than previously predicted. The BWRVIP did not recommend adjusting the recommended toughness for linear-elastic conditions.

Potential Impacts on Licensee Inspection Programs

Licensees determine inspection frequencies for RVI components through fracture mechanics evaluations of found or postulated flaws. As a result of the new information presented in the May 27, 2021 public meeting, the recommended fluence threshold for transitioning from ductile to brittle fracture in weld material was reduced. Furthermore, the BWRVIP indicated that the recommended J_R curve may be impacted. The fluence threshold impacts the failure mode assumption, while the change to J_R impacts the toughness assumption for ductile fracture. Therefore, inspection frequencies of welds calculated assuming ductile fracture may be increased as a result of the updated guidance, either through the new toughness assumption or through updating the failure mode assumption.

3.2 Operational History

Core shroud cracking in U.S. BWRs first came to light during the 1990's. The staff published NUREG-1544, "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components," (ADAMS Accession No. ML070300538) to describe the early operating experience and the actions taken by industry and NRC to disposition the issue. Chapter 5 of NUREG-1544 describes core shroud cracking at the Brunswick, Quad Cities, Dresden, and Vermont Yankee nuclear power plants. The most severe cracking reported was at the Brunswick H3 weld, with a 1.4 inch (in.) deep, 360° flaw. The industry has devised and implemented repair techniques for cracked core shrouds. For example, license renewal safety evaluation reports for the Hatch and Brunswick plants provide description of core shroud repair activity (see Section 3.1.15.2 under ADAMS Access No. ML020020291 and Section 4.2.9.1 under ADAMS Accession No. ML061730129).

The cracking mechanism is primarily attributed to intergranular stress corrosion cracking (IGSCC), with irradiation assisted stress corrosion cracking (IASCC) as another possibility. IGSCC may occur in areas of high cold work, whereas IASCC may be more dominant in areas subject to high neutron fluence. Investigations at the time suggested that the degradation was enhanced due to fabrication processes related to the plant-specific design. Current data available to the NRC staff suggests that modern cracking, which is more closely related to inservice degradation, is less severe than that observed in the 1990s.

Neutron exposure of the core shroud is plant-specific, as it depends on a number of design factors. The portion of the core shroud surrounding the fuel will receive the largest dose. Current information available to the NRC suggests that the fluence thresholds of BWRVIP-100 can play a role in licensee RVI inspection programs.

3.3 Flaw Analysis

Using the limited data available to the NRC, the staff performed a bounding allowable flaw size analysis as part of assessing the risk significance of the apparent nonconservatism of BWRVIP-100, Revision 1-A. The assumed inputs for the analysis are presented in Table 2.

Table 2: Inputs for Generic Flaw Size Analysis

Description	Value
Core Shroud Outer Diameter	207 in.
Core Shroud Thickness	1.5 in.
Stainless Steel Yield Strength	30 ksi
Stainless Steel Ultimate Tensile Strength	75 ksi
Applied Bending Moment	2×10^8 in.-lbs

The applied bending moment in Table 1 was taken from an analysis of a recirculation outlet line loss of coolant accident (LOCA).

The staff estimated allowable flaw sizes under limit load, elastic-plastic fracture mechanics (EPFM), and linear-elastic fracture mechanics (LEFM) failure modes. The allowable flaw length results for a through-wall circumferential flaw are shown in Table 3.

Table 3: Allowable Flaw Size Results

Failure Mode	Allowable Flaw Size, % of Circumference
Plastic Collapse (Limit Load Analysis)	75%
Ductile Tearing (EPFM Analysis)	70%
Brittle Fracture (LEFM Analysis)	18%

Furthermore, the staff researched potential flaw populations in the BWR fleet. The data comes from a variety of sources, including the Component Operational Experience, Degradation, and Aging Project event database, Generic Letter 94-03 and associated NUREG report NUREG-1544, and a pressurized water reactor (PWR) core barrel flaw analysis submitted to the NRC for review. Figure 1 shows the distribution of flaw sizes, based upon that data.

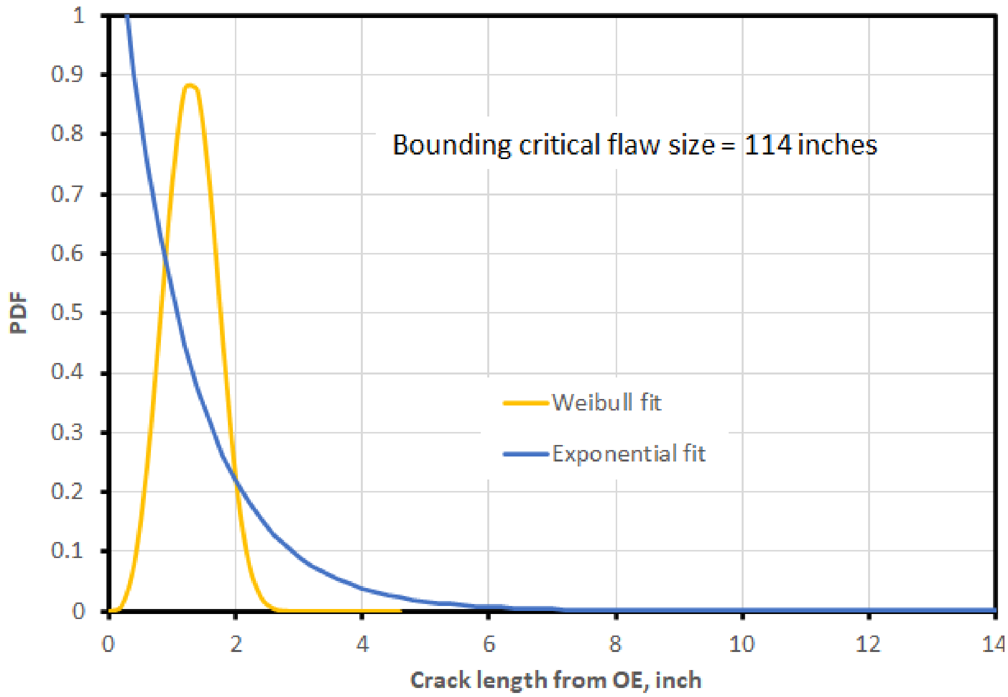

Figure 1: Estimated Flaw Distribution

These data illustrate that the probability of currently having a flaw equal to the critical flaw size (114 inches) is extremely small. Even in a low toughness material, brittle fracture will not occur in the absence of a flaw of critical length.

3.4 Risk Assessment

The staff evaluated the increase in risk by examining whether the nonconservatism in the flaw evaluation procedure of the core shroud could affect (i) frequency of the initiating events and (ii) probability of failure for the actions and components that are designed to mitigate accidents, and (iii) magnitude of consequences.

Potential Impact on Initiating Event Frequencies

The functions of the core shroud include directing reactor coolant around nuclear fuel and providing support for other core shroud region components consistent with the plant design. Therefore, a crack in the core shroud could cause a perturbation in flow and consequently result in a plant transient. No other initiators such as LOCAs or loss of offsite power would occur due to the core shroud failure. The CDF due to a general plant transient is typically about 1×10^{-6} / year. If one conservatively assumes a 10 percent increase of this occurrence due to the degraded core shroud, the conditional core damage frequency will remain in the order of 1×10^{-6} / year which is significantly less than the threshold of 1×10^{-3} / year.

The staff also noted that degradation of the core shroud could challenge core coolable geometry which may impact core flow inside the vessel. Such core flow issues may invalidate assumptions and analyses in the LOCA evaluations and various thermal hydraulic analyses. Based on the engineering analysis documented in Section 3.3 of this report, the probability of core shroud failure is extremely low and therefore a core coolable geometry is maintained and long-term heat removal would not be impeded even with the potential degradation of the core shroud.

Potential impact on accident mitigation failure probabilities

In terms of accident mitigations, the staff examined how the degradation of the core shroud may affect SSCs designed to mitigate accident sequences. The staff noted that the failure of vessel internal components could affect the reactor protection system that is designed to insert negative reactivity through the control rods to shut down the reactor. Based on the engineering analysis documented in Section 3.3 of this report, the currently known flaw distribution in the operating reactors chore shrouds is extremely small compared to the critical flaw size and therefore, the degraded condition of the core shroud has negligible impact on the control rods insertion. Therefore, the staff concluded that the degraded core shroud has an extremely low failure probability and therefore does not impact the SSCs designed to mitigate accidents.

Potential impact due to external hazards

In terms of the effect of external hazards, the staff determined that a seismic event may have the potential to contribute to core shroud degradation risk. In a previous LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergency Issues," for PWR baffle-former bolt (BFB) degradation (ADAMS Accession No. ML16225A341), the staff performed a simplified seismic risk evaluation. In the evaluation, a bounding seismic hazard curve and a generic seismic fragility value for RVI components were used and it was concluded that even with significant reduction in the seismic capacity of the RVIs due to degradation, the issue did not rise to the level of an imminent safety concern. The staff examined the assumptions in the BFB seismic evaluation and determined that the evaluation conclusion is applicable and remains valid for the degraded core shroud.

Risk Impact Conclusion

The staff examined how the potential degradation of the core shroud could affect the frequencies of the initiating events and the probabilities of failure for the actions and components that are designed to mitigate accidents. The staff concluded that the increase in risk associated with the BWRVIP-100, Revision 1-A nonconservatism is low.

4 Principles of Risk-Informed Decision Making

4.1 Compliance with Existing Regulations

The following discusses the relationship of the BWRVIP-100, Revision 1-A nonconservatism to each of these regulations.

10 CFR Part 50, General Design Criteria (GDC) for Nuclear Power Plants

The staff notes that the BWRVIP-100, Revision 1-A nonconservatism does not automatically result in noncompliance with the GDC below. Licensees are responsible for evaluating the impact of the new fracture toughness data on their RVI inspection program. If a licensee determines its plant is in noncompliance with any of the Commission's regulations, including the GDC, the licensee is responsible for taking appropriate corrective action, which may include shutting down the plant.

GDC 4 Environmental and dynamic effects design bases.

The RVI were designed to accommodate these environmental effects, such as loads from LOCAs and seismic events. The staff's bounding allowable flaw size calculation in Section 3.3 was based upon LOCA loads. Normal operating loads on the core shroud are trivial. Even in the most limiting failure condition (brittle fracture), the staff's analysis suggests that the core shroud can tolerate a through-wall circumferential flaw that extends 18 percent of the circumference during a LOCA. Therefore, core shroud degradation does not automatically indicate noncompliance with GDC 4.

GDC 27 Combined reactivity control systems capability.

If an accident such as a LOCA occurs, failure of the core shroud could challenge safe shutdown capability. However, licensee inspection programs are designed to ensure that the critical flaw size is not reached during a given timeframe. Therefore, BWRVIP-100, Revision 1-A nonconservatism does not automatically indicate noncompliance with GDC 27. Licensees must, however, appropriately update their inspection programs to account for earlier transition to brittle fracture and decreased toughness in the weld material.

10 CFR 50.46 and GDC 35

If an accident such as a LOCA occurs, failure of the core shroud could challenge core coolability. However, licensee inspection programs are designed to ensure that the critical flaw size is not reached during a given timeframe. Therefore, BWRVIP-100, Revision 1-A nonconservatism does not automatically indicate noncompliance with 10 CFR 50.46 and GDC 35. Licensees must, however, appropriately update their inspection programs to account for earlier transition to brittle fracture and decreased toughness in the weld material.

10 CFR 50.55a

BWR core shrouds fall under Section XI, Category B-N-2, Item Number B13.40, which requires VT-3 examination once each 10-year inspection interval. However, licensees may also implement the inspection program described in BWRVIP-76, Revision 1-A. Inspection frequencies calculated under BWRVIP-76, Revision 1-A procedures will likely be impacted by the BWRVIP-100, Revision 1-A nonconservatism.

10 CFR 54.21

Licensees rely on inspection programs to ensure that the RVIs will be maintained consistent with the current licensing basis for the period of extended operation. While the presence of BWRVIP-100, Revision 1-A nonconservatism does not imply noncompliance with 10 CFR 54.21, licensees must appropriately evaluate the impact of the new toughness data on their inspection programs.

4.2 Consistency with the Defense-in-Depth Philosophy

One factor for assessing how an event might degrade defense in depth is to see how it affects the balance among the layers of defense. It is useful to consider the following layers of defense (successive measures) when evaluating the impact of the BWRVIP-100, Revision 1-A nonconservatism on defense-in-depth:

1. a robust plant design to survive hazards and minimize challenges that could result in an event occurring,

2. prevention of a severe accident (core damage) should an event occur,

3. containment of the source term should a severe accident occur, and

4. protection of the public from any releases of radioactive material (e.g., through siting in low-population areas and the ability to shelter or evacuate people, if necessary).

Restricting the discussion to BWR core shrouds, the most severe scenario is failure of the shroud in a manner that impacts safe shutdown capability or core coolability. Normal operating loads on the core shroud are very low, given that the core shroud is not a pressure-retaining component. Therefore, the structural stability of the core shroud is expected to be challenged only under accident conditions. The staff's bounding critical flaw size analysis in Section 3.3 indicated that the core shroud can tolerate relatively severe cracking, even under LOCA loads. Such cracking is likely managed by the licensees at this point through repair activities. Therefore, it is unlikely that such a severe flaw is currently in service, such that a LOCA will fail the core shroud and increase consequences of the event. Furthermore, the simultaneous occurrence of the severe flaw and the LOCA event is highly unlikely.

The new toughness information does not affect the containment structure, such that the containment response to accident conditions would not be impacted. Finally, the BWRVIP-100, Revision 1-A nonconservatism has no effect on the emergency preparedness functions, such that the fourth layer of defense is not affected.

Another factor of defense in depth to consider involves the effect of the issue on the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment). As stated above, core shroud performance does not affect the containment response. Core shroud failure during a LOCA event could affect fuel cladding in peripheral fuel assemblies as a result of contacting surfaces (e.g., wear). This would affect only a small percentage of the cladding. Therefore, core shroud may result in a small increase the consequences of an accident, but it does not impact the likelihood of an accident occurring.

In summary, the BWRVIP-100, Revision 1-A nonconservatism does not affect the containment function. The effect on cladding is small and limited to peripheral fuel assemblies. The BWRVIP-100, Revision 1-A nonconservatism does not indicate that an imminent safety concern exists from any effect on defense in depth, because it does not significantly affect the four layers of defense nor the containment fission product barrier.

4.3 <u>Maintenance of Adequate Safety Margins</u>

Failure of the core shroud during normal operation is a highly unlikely event since the loads on the shroud are so small. Licensee RVI inspection programs are in place to detect cracking and establish appropriate inspection frequencies to protect against shroud failure under accident conditions. Therefore, the staff concludes that there is very little impact on safety margins as a result of the BWRVIP-100, Revision 1-A nonconservatism. However, licensee RVI inspection programs should be updated to account for the new information on toughness and failure mode.

4.4 <u>Demonstration of Acceptable Levels of Risk</u>

The staff evaluated the increase in risk due to the potential degradation of the core shroud in Section 3.4. In the evaluation, the staff noted that the currently-known flaw distribution in the operating reactors chore shrouds is extremely small compared to the critical flaw size. Therefore, the probability of core shroud failure is extremely low. The staff examined how the potential degradation of the core shroud could affect the frequency of the initiating events and the probability of failure for the actions and components that are designed to mitigate accidents. The staff concluded that the increase in risk associated with the BWRVIP-100, Revision 1-A nonconservatism is low.

4.5 <u>Implementation of Performance Monitoring Strategies</u>

Licensees have the capability to update their RVI inspection programs as new information becomes available. The staff may monitor plant-specific impacts of the BWRVIP nonconservatism through the reactor oversight or similar process. Relevant plant-specific information includes:

1. fluence maps on the core shroud inner and outer surfaces,
2. flaw size and location distributions,
3. inspection frequency prior to the new toughness information, and
4. inspection frequency after the new toughness information.

While analysis of the other key principles suggests that no immediate safety concern exists, NRC action to provide independent verification of licensee corrective actions may be prudent in order to support the principle of performance monitoring.

5 Option Description and Evaluation

To address the nonconservatism in BWRVIP-100, Revision 1-A, the staff considered three options. Table 4 lists the options and describes the staff's evaluation criteria for each option. In the following paragraphs, each of these options is first described briefly and then evaluated with respect to the five principles for risk-informed decisions.

Table 4: Staff Options and Evaluation Criteria

#	Option	Evaluation Criteria
1	Issue Shutdown Orders	This action would be required if an imminent safety concern were identified, such as: • Defense in depth is significantly degraded • There is significant loss of safety margin • Increase in CDF is greater than or on the order of 10^{-3} /year or increase in LERF is greater than or on the order of 10^{-4} /year
2	Leverage Reactor Oversight Program	This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines: • Adequate defense-in-depth is maintained • Sufficient safety margin is maintained • An acceptable level of risk is maintained • Additional information is needed to establish that the aforementioned assessments have an adequate degree of conservatism • Additional information is needed to make a regulatory decision
3	Take No Action	This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines: • Adequate defense-in-depth is maintained • Sufficient safety margin is maintained • An acceptable level of risk is maintained • The adequacy of defense-in-depth, safety margin, and risk level have a degree of conservatism that provides reasonable assurance that the potential safety impact of the nonconservative toughness (or, failure mode) is bounded

5.1 Option 1: Issue Shutdown Orders

Synopsis: This option consists of shutting down some or all operating reactors through a regulatory process (such as an order) until inspections, analyses, and mitigation are conducted to provide reasonable assurance that the calculated risk levels are acceptable. This option is preferable if there is an immediate safety issue such that the risk to operating plants is clearly demonstrated to be large and immediate. LIC-504 defines such risk as a Conditional CDF greater than 1×10^{-3} /year or Conditional LERF greater than 1×10^{-4} /year.

Principle 1: Immediate shutdown and inspection would identify the latest flaw populations, and licensees would be required to update their RVI inspection programs to account for the new fracture toughness data. With the updated inspection programs, compliance with the regulations would be ensured.

Principle 2: Option 1 is consistent with the defense-in-depth philosophy because it would ensure flaw populations are acceptable and RVI inspection programs are updated before startup. Thus, the likelihood of core damage is minimized with this option and all three barriers to fission product release (fuel cladding, reactor coolant pressure boundary, containment) are intact.

Principle 3: While the safety margins are not currently impacted by the BWRVIP-100 nonconservatism, immediate shutdown for inspection would determine the extent of degradation. Corrective actions would provide further demonstration of adequate safety margins by repair or flaw evaluation. Therefore, Option 1 would ensure that adequate safety margins are maintained.

Principle 4: The bounding risk assessment of operating with core shroud analyzed under original BWRVIP-100, Revision 1-A guidelines indicates that this issue does not rise to the level of an imminent safety concern because the estimated increase in CDF is less than 1×10^{-3} per year, as detailed previously. The level of risk for Option 1 would represent no change from the condition of having uncracked core shroud because the susceptible plants would be shut down immediately.

Principle 5: Implementation of Option 1 would allow immediate inspection and mitigation, which are the most effective performance monitoring strategies for core shroud cracking.

5.2 Option 2: Leverage NRC Inspection Programs

Synopsis: EPRI has initiated communications with impacted licensees, describing the potential nonconservatism and how licensees may need to adjust their plant-specific programs. This option leverages existing Regional inspection programs to target licensee corrective actions to address the BWRVIP-100, Revision 1-A nonconservatism.

Principle 1: Inspection activities provide an independent check on licensee corrective actions. Verification of licensee activities responding to emerging information provides reasonable assurance of compliance with the regulations.

Principle 2: Option 2 is consistent with the defense-in-depth philosophy because core shroud degradation does not directly impact the three barriers to fission product release (fuel cladding, reactor coolant pressure boundary, containment). Thus, the likelihood of core damage is minimized with this option and all three barriers to fission product release are intact.

Principle 3: Option 2 is consistent with the maintaining safety margins because it would ensure licensee inspection frequencies of RVI components are sufficiently conservative. Corrective actions would provide further demonstration of adequate safety margins in updated flaw evaluations. Therefore, Option 2 would ensure that adequate safety margins are maintained.

Principle 4: The risk assessment of operating with core shroud analyzed under original BWRVIP-100, Revision 1-A guidelines indicates that this issue does not rise to the level of an imminent safety concern because the estimated increase in CDF is less than 1×10^{-3} per year, as detailed previously. Degradation occurs slowly over time, so Option 2 is judged acceptable from a risk perspective.

Principle 5: Implementation of Option 2 would provide independent verification that licensee inspection programs remain current as new information is discovered. Therefore, the staff would have reasonable assurance of effective performance monitoring.

5.3 Option 3: Take No Action

Synopsis: In this option, the agency takes no programmatic approach to verify licensee corrective actions. Headquarters staff may still ask appropriate requests for additional information on licensing actions related to the BWRVIP-100, Revision 1-A nonconservatism. This option entirely relies on proper execution of licensees' corrective action programs.

Principle 1: The staff does not have independent verification of licensee corrective actions to address BWRVIP-100 nonconservatism. Therefore, the staff cannot draw informed conclusions about licensee compliance.

Principle 2: Option 3 is consistent with the defense-in-depth philosophy because core shroud degradation does not directly impact the three barriers to fission product release (fuel cladding, reactor coolant pressure boundary, containment). Thus, the likelihood of core damage is minimized with this option and all three barriers to fission product release are intact.

Principle 3: Option 3 is consistent with maintaining safety margins. The operating experience indicates that the industry implemented appropriate corrective actions to address the relatively severe core shroud cracking discovered in the 1990s. All data available to the staff suggests that modern core shroud cracking is much less severe, such that the critical flaw size is not challenged even under LOCA conditions. Therefore, the staff has reasonable assurance that safety margins are maintained under Option 3.

Principle 4: The bounding risk assessment of operating with core shroud analyzed under original BWRVIP-100, Revision 1-A guidelines indicates that this issue does not rise to the level of an imminent safety concern. However, it is not clear how long susceptible plants would operate before analyzing the potential impacts and how subsequent inspections would be defined. Additional time to resolve this issue increases the uncertainty with how this issue might progress in the future and also increases the associated level of risk. Therefore, the increase in CDF may not be acceptable for longer term operation.

Principle 5: Implementation of Option 3 does not provide independent verification that licensee inspection programs remain current as new information is discovered. Therefore, under Option 3, the staff does not have reasonable assurance of effective performance monitoring.

6 Recommendations

The staff recommends Option 2. Based upon the discussion in Section 5, Option 2 provides appropriate oversight of licensee corrective actions without negatively impacting NRC and industry resources. Focused inspection activities within the Reactor Oversight Process (or some similar process) will provide information to the staff about plant-specific RVI inspection programs and will serve as an independent check on licensee activities related to the BWRVIP-100, Revision 1-A nonconservatism.

Option 1 is not justified in light of the low risk associated with the BWRVIP-100, Revision 1-A nonconservatism. While failure of the core shroud might impact safe shutdown capability, the low loads on the shroud during normal operation make core shroud failure a highly unlikely event. Significant bending load can result from LOCA conditions, but then a relatively severe flaw must be present in the core shroud concurrent with the unlikely LOCA event. These factors lead to the low risk associated with the BWRVIP-100, Revision 1-A nonconservatism. Therefore, the staff judges that immediate regulatory action is unwarranted.

While Option 3 has the least impact on NRC and industry resources, it also provides no independent oversight of plant-specific programs. The NRC would be entirely relying on the interactions with EPRI and the limited information available to staff, without observing actual licensee corrective actions and drawing independent conclusions about their efficacy. Therefore, the staff does not recommend Option 3.