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Containment Reliability and Risk-Informed Decision Making–A Perspective

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ABSTRACT

For the light-water reactors in the United States, the containment structure and the associated pressure retaining components are the last barrier to the release of radioactive materials to the environment during the postulated accident conditions. The reliability of the containment components to perform their intended function is verified by ensuring that the accessible portions of the containment components are periodically inspected and maintained. Additionally, the leak-tightness of the containment is verified by periodic leak rate testing of the containment structure and its pressure retaining components.

Since the 1979 accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) staff has been studying various severe accident scenarios and gradually incorporating the use of probabilistic risk assessment (PRA) in the regulatory process. In 1988, the NRC staff requested information on the assessment of severe accident vulnerabilities from the operators of nuclear reactors. By the mid-1990s, in conjunction with the responses to these requests and further research by the NRC staff in the use of the PRA methodologies, the NRC felt confident PRA could be used in the regulatory process.

The U.S. nuclear industry is using risk-informed methodologies in reducing unnecessary regulatory burden on the operators of the nuclear reactors and in addressing various issues of concern to the industry. One of the issues addressed by the industry is increasing the time-interval for performing the leak rate tests of containment components using a risk-informed approach. The American Society of Mechanical Engineers is also developing risk-informed requirements for periodic inspection of containment components.

This paper discusses the critical factors that need to be considered in risk-informed decision making to ensure containment reliability.

KEY WORDS: containment, reliability, risk-informed, design basis accident, severe accident, inservice inspection, leak rate testing, Nuclear Regulatory Commission (NRC)

INTRODUCTION

The intended function of containment is defined in Criterion 16 of Appendix A to 10 CFR Part 50 (Ref. 1) of the NRC regulations as follows:

"Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require." Additionally, General Design Criterion 2, "Design bases for protection against natural phenomena," requires the containment to remain functional under the effects of postulated natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches.

Thus, in a relatively narrow definition of "reliability," the reliability of a containment of light water-cooled reactors can be conceived as its success probability of withstanding the postulated accident conditions and natural phenomena. A typical light water reactor containment consists of a containment structure, over 100 electrical and mechanical penetrations, two to three equipment access hatches, and personnel air locks. The pipes in the mechanical penetrations are likely to carry radiological products during normal operation or postulated accident conditions, and have isolation provisions, so that the containment could be isolated during a radiological accident. In the rest of this paper, "containment" is synonymous with a containment system comprising the containment structure, penetrations, and isolation valves. Containment is a vital engineered safety feature of a nuclear power plant. In normal operating conditions, the containment is subjected to operating and environmental stressors such as ambient pressure fluctuations, temperature variations, earthquakes, ice, and windstorms. In some containment designs, the principal leak-tight barrier is surrounded by another structure, such as a shield wall or a shield building, which protects the containment from some of the external influences, and in some cases collect leakage from the primary containment for processing prior to release to the atmosphere. The mechanical stresses and strains generated by transients under normal conditions and the effects of high-probability ($>10^{-2}$) external influences are a small fraction of the limiting conditions for which the containment is designed. However, the fatigue life of the containment can be affected by the significant number of cycles of such low-stress transients. The containment is also subjected to various types of internal degradation (aging degradation), depending on the inherent characteristics of the materials, the fabrication processes, and the construction methods. The rate and extent of the degradation are influenced by sustained environmental conditions such as temperature, humidity, water leakage, and borated water spills.

Thus, the reliability (or the success probability) of a containment to perform its intended function under design basis conditions as well as under higher loads due to severe accidents and earthquakes is influenced by the containment=s inherent capability and the various stresses and degradation mechanisms that act on it. The basic concept in ensuring the reliability of the containment is to track the degradation of the containment components through periodic inspections and check the leak-tight integrity of the containment's pressure-retaining components through periodic leak rate testing

DETERMINISTIC APPROACH

As implemented in the United States, the deterministic approach consists of conducting (1) periodic inspection of the containment pressure-boundary components, and (2) leak rate testing of the containment components. The basic features of the inspection and leak rate testing are discussed in the following paragraphs.

Until 1996, there was no formal program for inspecting containments, except for the inspection of prestressing tendons in prestressed concrete containments. For prestressed concrete containments with unbonded tendons (where the prestressing elements and associated anchorage components are protected from corrosion by a specially formulated grease), the monitoring of the prestressing system was performed by the implementation of Regulatory Guide (RG) 1.35 (Ref. 2). However, in the early 1990s, the NRC learned of a number of incidents of significant degradation of the prestressing tendon system, steel liner of concrete containment structures, and steel shells of steel containments. The NRC documented the major findings in NUREG-1522 (Ref. 3). The findings summarized in Reference 3 were identified by containment tendon inspections, by ad hoc inspections by the plant personnel, NRC inspectors, or, to a lesser degree, by inspections required before the leak rate testing of the containments pursuant to 10 CFR Part 50, Appendix J (Ref. 4). Even when degradation was found, there was no formal procedure for documenting the degradations and implementing corrective actions.

In August 1996, the NRC amended its regulation 10 CFR 50.55a (Ref. 5) to incorporate by reference the 1992 Edition and the 1992 Addenda of Subsections IWE (Ref. 6) and IWL (Ref. 7) of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for inspection of containment structures in light-water-cooled nuclear power plants, with certain additions and modifications. Subsection IWE provides the requirements for inservice inspection, repair, and replacement of Class MC pressure-retaining components (i.e., steel containments) and their integral attachments, and metallic shell and penetration liners of Class CC pressure retaining components (i.e., concrete containments) and their integral attachments. Subsection IWL provides the requirements for pre-service examination, inservice inspection, and repair of the reinforced concrete and post-tensioning systems of Class CC components. In conjunction with the additional requirements in Reference 5, a general visual examination of the accessible portion of the entire containment is required to be performed 3 times in 10 years.

Reference 4 entitled, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," establishes the testing requirements for preoperational and periodic verification of the leak-tight integrity of the primary reactor containment, including systems and components which penetrate the containments of light water-cooled power reactors, and establishes the acceptance criteria for such tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment does not exceed allowable leakage rate values as specified in the technical specifications, or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are done during the service life of the containment, and systems and components penetrating the containment.

The reactor containment leakage test program includes the performance of integrated leak rate testing (ILRT), also known as Type A test, containment penetration leak rate testing (Type B), and Containment isolation valve leak rate testing (Type C). The Type B and Type C tests are also referred to as local leak rate tests (LLRTs). The current

Appendix J requirements provide two options for performing the tests. Option A (deterministic), and Option B (performance-based).

Option A (deterministic) requires that after the preoperational leakage rate tests, three integrated leak rate tests (Type A tests) be performed at approximately equal intervals during each 10-year service period. Option A requires Type B tests to be performed during reactor shutdown or refueling, but in no case at intervals greater than 2 years. For containments employing continuous leakage monitoring, Option A requires the Type B tests to be performed every other reactor shutdown for refueling or every 3 years, whichever is less. Air locks are required to be tested every 6 months. Option A requires Type C testing to be performed every refueling outage, or every 2 years, whichever is less.

The NRC considers that the implementation of these deterministic requirements, that is, periodic inspection of accessible areas, and periodic leak rate testing of containment components, will maintain the high reliability (in the qualitative sense) of the containment.

PERFORMANCE-BASED APPROACH

Some intuitive performance-based criteria for inspection of containment structures are built into the existing codes (i.e. Subsections IWE and IWL of Section XI of the ASME Code). For example, in IWE-2420(c): “When the reexaminations required by IWE-2420(b) reveal that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, the areas containing such flaws, repairs or degradation, no longer require augmented examination.” Such logic is found in all prescriptive codes. There is no quantitative basis for the criterion to conclude that the conditions observed during the three consecutive inspection periods would represent the benign nature of the degradation (it could be two or four). However, the criterion is based on the intuitive judgment of the knowledgeable committee members. In many cases, this kind of consensus approach is appropriate in setting prescriptive requirements.

However, the nuclear industry had expressed its concern that the deterministic prescriptive requirements are too costly to the owners of nuclear power plants, and that steps could be taken to reduce the burden on the industry. As a result, NRC began looking at the costs and benefits of implementing the Option A requirements of Appendix J.

Earlier NRC studies documented in NUREG-1273 (Ref. 8) had indicated that the public risk associated with undetected containment leakage as a result of implementing Option A of Appendix J, is very small. These studies recommended that the Type A, Type B, and Type C tests required by Option A should be continued since they provide the assurance of the continued high availability of the containment to perform its intended function. In the early 1990s, in conjunction with the database collected by the Nuclear Energy Institute (NEI), NRC attempted to quantify the risk associated with increasing the time interval for performing ILRT. This study is documented in NUREG-1493 (Ref. 9).

NUREG-1493 analyzed the risk data presented in NUREG-1150 (Ref. 10) for five plants: Surry, Peach Bottom, Sequoyah, Grand Gulf, and Zion. The NUREG-1150 risk data was made applicable to the focus of the report, i.e. to provide technical bases for revising the Appendix J requirements. Additionally, NUREG-1493 analyzed the operating data from North Anna and Grand Gulf Plants. Based on the operating data, the study calculated the change in risk (in person-rem) to the public for the 15 alternatives considered in the study. For example, Alternative 4 maintains the current Appendix J acceptance criteria for Type A, Type B, and Type C tests, and reduces the ILRT frequency from three per 10 years to one per 10 years. The study concluded that the cost saving to the industry would be around $500 millions, with an insignificant increase in the risk to the public.

Option B of Appendix J does not provide prescriptive requirements for scheduling the periodic ILRTs and LLRTs. It simply states: “Type A test must be conducted (1) after the containment system has been completed and is ready for operation and (2) at a periodic interval based on the historical performance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents.” For Type B and Type C tests, Option B states: “Type B pneumatic tests to detect and measure local leakage rates across pressure retaining, leakage-limiting boundaries, and Type C pneumatic tests to measure containment isolation valve leakage rates, must be conducted (1) prior to initial criticality, and (2) periodically thereafter at intervals based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release to reduce the risk from reactor accidents.”

NEI 94-01 (Ref. 11) contains the industry guidelines for implementing the Option B requirements. The report briefly discusses the Appendix J, Option B requirements and current testing methodologies for Type A, Type B, and Type C testing, and provides guidelines for determining the performance-based tests intervals for the three types of tests. For performing Type A, Type B, and Type C tests, the report recommends the use of ANSI/ANS 56.8 (Ref. 12). The report principally relies on the risk assessment in NUREG-1493 for establishing the guidelines. In Sections 9 and

10, the report provides guidelines for establishing test intervals for Type A, Type B, and Type C tests. The following is a brief description of the relevant guidelines for Type A, Type B, and Type C tests.

Type A Tests

A preoperational test shall be performed prior to reactor initial operation, and subsequent periodic Type A tests shall be performed within 48 months of the prior successful Type A test. The periodic Type A test interval can be increased to 10 years based on an acceptable performance history. An acceptable performance history is defined as completion of two consecutive periodic Type A tests where the "calculated performance leakage rate" was less than 1.0 L_a (where L_a is the acceptable leakage rate expressed in percentage of weight of dry air in the containment) at the containment peak calculated pressure for the design basis accident. The report provides guidance for calculating the performance leak rate. It also establishes an acceptable criterion of $0.75L_a$ for as-left Type A leakage rate tests. Thus, the acceptance criteria for assessing the results of Type A tests are essentially the same as those required by Option A of Appendix J.

Section 9.2.6 of the report provides guidelines for corrective actions, if the as-found Type A performance leakage rate is not acceptable. The report states, "Once the cause determination and corrective actions have been completed, acceptable performance should be reestablished by performing a Type A test within 48 months following the unsuccessful Type A test." Furthermore, the report recommends that if the unacceptable Type A performance leakage rate is due to the failure of Type B or Type C test components, the unsuccessful test need not be included in the determination of the Type A performance leakage rate and Type A test interval. The corrective actions should follow the Type B and Type C corrective action guidelines.

Type B Tests

Consistent with the current practice, Type B test interval criteria are established separately for containment air locks and other penetrations.

For containment air locks, preoperational test shall be performed prior to the preoperational Type A test, and subsequent periodic testing of air lock shall be performed once every 24 months. The air lock components, such as equalizing valves, door seals, and resilient seals shall be tested once per every 24 months. The report refers to ANSI/ANS 56.8 for procedures for testing air locks and their components. Sections 10.2.2.2 and 10.2.2.3 recommend generic procedures for repair and adjustment of air locks and for corrective actions.

For other penetrations, Type B tests shall be performed prior to initial reactor operation. Subsequent periodic Type B tests shall be performed at a frequency of at least once every 24 months, until acceptable performance is established as defined in Section 10.2.1.2 of the report. The report allows up to 10 years between Type B tests, based on the performance of the individual penetration.

Type C Tests

NEI 94-01 recommends that Type C tests be performed prior to initial reactor operation. Subsequent Type C tests shall be performed at a frequency of at least 24 months. Test intervals of Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests, where the results of each Type C test are within a licensee's allowable administrative limits. When the Type C test performance criteria are met, the report permits the Type C test interval to be as much as 10 years.

Regulatory Guide 1.163 (Ref. 13) endorses the use of NEI 94-01 and ANSI/ANS-56.8 for implementing the performance-based option of Appendix J with certain exceptions, such as that the maximum Type C test interval is 5 years instead of the 10 years recommended in NEI 94-01.

RISK-INFORMED APPROACH

From the standpoint of time and resources, the integrated leak rate testing of containment is costly. The objective of making the tests performance-based was to help licensees reduce the refueling outage time while maintaining an adequate level of safety. The objective can be accomplished if the plant-specific performance as well as the industry-wide data shows insignificant increase in risk from increasing the interval between Type A tests. NUREG-1493 showed an insignificant increase in risk by extending the Type A test time interval to as much as 20 years. To help licensees performing plant-specific risk assessments, Electric Power and Research Institute (EPRI) developed a methodology to assess the risk impact of revising the ILRT intervals. The methodology is described in EPRI TR-104285 (Ref. 14).

The EPRI report uses the basic methodology for risk assessment from NUREG-1493, but systematizes the steps to help licensees integrate their plant-specific risk assessment results for the relevant risk analysis. NUREG-1493 utilized the NUREG-1150 population dose model to estimate the risk.

When the NUREG and EPRI reports were developed (1993-1994), the NRC had not developed explicit guidelines for changing the licensing basis (LB) using risk-informed decision making. In 1998, the NRC issued RG 1.174 (Ref. 15), which provides quantitative guidelines for making licensing basis changes. The criteria are established based on the Commission Final Policy Statement for Probabilistic Risk Assessment (PRA), which establishes the safety goals for nuclear power plants as follows:

"Through the Safety Goal Policy Statement, the Commission propagated its philosophy that the risk from the operation of a nuclear power plant should be no more than 0.1 percent of the risk to which people are exposed from other sources. This statement of risk translates into objectives on individual risk of 2E-06/yr of a latent fatality and 5E-07/yr of an early fatality." The subsidiary numerical objectives to be used with appropriate consideration of uncertainties in making regulatory judgments are: (1) a core damage frequency (CDF) of 1E-04/reactor-year (RY) as for accident prevention, and, (2) a conditional containment failure probability (CCFP) of 0.1 for accident mitigation.

In its policy statement, the Commission also states: "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

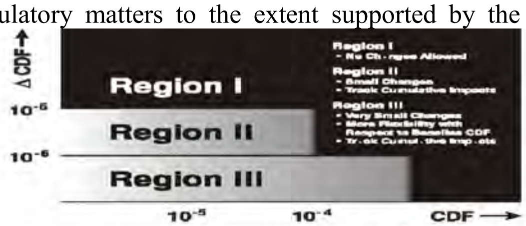

Figure 3. Acceptance Guidelines* for Core Damage Frequency (CDF)

Figure 4 Acceptance Guidelines* for Large Early Release Frequency (LERF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decisionmaking, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

1.174-16

Based on this philosophy, the NRC staff developed RG 1.174. The acceptance criteria to be used for making licensing basis changes are reproduced in Figures 3 and 4.

In the guide, CDF and large early release frequency (LERF) are considered sufficient for assessing the safety of a nuclear power plant, and changes in CDF (i.e. ΔCDF) and LERF (i.e. ΔLERF) are sufficient for assessing the impact of the change in the currently established requirements on the safety of the plant. The guide provides a number of qualitative pointers on PRA quality assurance and how to account for uncertainties. It also permits the NRC staff to base its safety assessment on the risk-informed criteria in combination with deterministic assessment.

Recognizing the low risk significance of increasing the ILRT intervals, a number of licensees performed risk assessments and submitted requests for approval of one-time licensing basis changes to extend the interval for performing the ILRT to up to 15 years. The requests included risk assessments based on the techniques of EPRI report and RG 1.174 criteria. It should be noted that for ILRT extension requests, the ΔLERF criterion governs the risk-informed decision making. The NRC staff has been reviewing and approving such requests on a case by case basis. The industry is also developing generic methodologies to obtain approval for a group of similar plants to change the containment ILRT interval to 20 years.

DISCUSSION

Potential uncertainties are an important consideration in risk-informed decision making.. Inservice inspections (ISIs) and leakage rate tests (LRTs) reduces the uncertainties related to the containment's ability to perform its intended function. Since the ISIs and LRTs performed in accordance with the prescriptive deterministic approach are the current requirements for assuring a qualitatively high containment reliability, the impact of reducing the frequency of performing the ISIs and LRTs needs to be addressed in relation to the increase in CDF and LERF, as discussed above. A number of combinations of ISI frequencies, and LRT frequencies are possible. However, thus far, the NRC staff has received risk-informed applications for reducing the frequency of the ILRT, keeping the same performance-based LLRT frequencies and deterministic ISI frequencies. The following is a summary description of the factors involved in approving such applications.

The applications submitted by the nuclear power plant (NPP) licensees included a discussion of the risk consequences of extending the ILRT frequencies from three times in 10 years to once in 10 and 15 years. In assessing the risk impact of extended ILRT test intervals, the licensees assess the impact of change on the large early release frequency, the off-site consequences (person-rem per year), and the conditional containment failure probability. Furthermore, the licensees stipulated that the containment structural integrity would be maintained between the requested time intervals through the ISI of containments performed in accordance with the requirements of 10 CFR 50.55a. During the in-depth reviews of the proposed licensee requests, the NRC staff pointed out that the containment ISIs and ILRTs supplement each other in ensuring the integrity of containments. Hence, the staff requested plant-specific information on containment ISIs to determine whether extending the ILRT interval would not adversely affect the containment integrity.

Over the last 15 years, a number of age-related degradation events have been identified. These included the corrosion degradation of a steel shell of a Mark I containment and the steel liners of both a Mark I and two dry pressurized-water reactor containments. The information gathered in Reference 3, and additional information reported since then, convinced the NRC staff, that any attempt to reduce the frequency of ILRTs through risk-informed decision making must incorporate the potential for such degradation in the risk assessment. During the reviews of such applications, one of the generic questions raised by the NRC staff was related to the consequences of degradations that could occur in the uninspectable (e.g., embedded) areas of the containment. The NRC staff pointed out that such degradations could not be detected by the visual examination performed in accordance with Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code. Accordingly, as part of the technical reviews of ILRT interval extension requests, the NRC staff sought information on how the potential leakage due to the age-related degradation mechanisms described above is factored into the risk assessment for the extension of the ILRT interval.

Neither NUREG-1493, EPRI TR-104285, nor NEI 94-01 provides guidelines for incorporating the effects of degraded containments in licensees' risk assessments. The periodic containment ISIs cannot detect the existence of corrosion on the uninspectable side of containment. If such concealed corrosion has become widespread, an ILRT performed at peak design pressure could show unacceptable leakage through the containment. Such degradations at aging nuclear power plants, were discovered at Oyster Creek in1991, at Brunswick in-1999, at North Anna in1999 and at D.C. Cook in 2000, and can potentially occur at other plants in the future. It is imperative to address the potential for such concealed degradations in the risk assessment related to the ILRT intervals.

A credible risk-informed analysis requires a credible database. The database should reflect the number of ILRT failures that have occurred in the past, with a stipulation that these failures would have gone undetected, if the ILRTs were not performed as required. However, the database cannot be directly used without some modifications for uncertainties and unidentified degradations of containment components.

It should be noted that the databases discussed in Chapter 4 of NUREG 1493 and used for the statistical analysis were obtained from plants which were relatively new at that time (less than 15 years old) and the ILRT failure data were obtained when the tests were performed at higher frequencies (i.e. three times in 10 years). Unless licensees request a one-time extension to perform ILRT at more than 10 years, the ILRTs will be performed at 10 years. With plants experiencing environmental and aging degradations as they age, the ILRTs for the plants tested in "as is" condition are likely to have higher failure rates than the ones analyzed in NUREG-1493. Depending upon the timing of performing the Type B tests, in relation to the ILRT test, the consequence of relaxing the test interval for Type B tests is likely to result in higher ILRT failure rates.

Thus, to make a credible risk-informed decision on the frequency of containment ILRT, the following factors should be considered:

1. The plant-specific ILRT data should contain all information needed for risk-analysis. For example, if during the performance of a Type A test, the leak rate is found excessive at very low pressure, but after some corrective action, the repeated test is found acceptable, the test may be considered as a failed ILRT, as the containment would not have satisfied General Design Criterion 16 (Ref. 1) in the as-found condition.
2. The industry-wide database should be consistent and should include all operating plants. For example, Table Summary 1 of Reference 9 should be based on actual values from all plants rather than from a range of values from a few selected plants.
3. As discussed above, potential degradation on the uninspectable side of the containment should be factored in the risk analysis..
4. The plant-specific probabilistic risk assessment should be updated (as needed) to reflect the latest industry-wide database on the degradation of vital components.

SUMMARY AND CONCLUSION

Containment reliability can be assured by proper inspection and performance testing of containment components. When the judgment regarding the containment's reliability is based on risk-informed assessment, all factors that contribute to its structural and leak-tight integrity have to be considered in the risk assessment. It is feasible to develop a probabilistic model which would consider frequencies of performing inspections and leak rate testing in relation to the resources required to conduct such inspections and tests, keeping the conditional probability of containment failure to less than 0.1. Since risk-informed methodologies are used to analyze the response of the containment to severe accidents, it is important to consider accident sequences that address the containment's vulnerability to degradations which cannot be detected by visual examination methods. At present, in the United States reliance has been placed on combination of risk-informed assessment backed with deterministic analysis to assess the effects of containment degradation on containment's reliability.

When decisions on making changes in the requirements are based on risk assessment, Section 2.3 (Element 3) of RG 1.174 (Ref.15) states, "Careful consideration should be given to implementation and performance-monitoring strategies. The primary goal for this element is to ensure that no adverse safety degradation occurs because of the changes to the licensing basis. The staff's principal concern is the possibility that the aggregate impact of changes that affect a large class of structures, systems, and components (SSCs) could lead to an unacceptable increase in the number of failures from unanticipated degradation, including possible increases in common cause mechanisms. Therefore, an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of SSCs that have been evaluated. This will ensure that the conclusions that have been drawn from the evaluation remain valid." Thus, in the context of the containment reliability, a careful consideration should be given to implementation and performance-monitoring strategies to ensure that no adverse consequences occurs because of the simultaneous relaxation of ILRT, LLRT, and ISI intervals.

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