Enclosure 4

RISK INFORMED TECHNICAL SPECIFICATION TASK FORCE (RITSTF)
RISK MANAGEMENT TECHNICAL SPECIFICATION INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
1	Technical Specification Required Actions Preferred End States	• NRC provided comments on TSTF-422 on 6/23/03. TSTF and RITSTF provided responses on 11/10/03. • TSTF-423 submitted to NRC on 8/12/03.	• TSTF-422, Rev. 1 was submitted to the NRC on 1/24/03. • The NRC provided comments on TSTF-422 on 6/23/03. The TSTF and RITSTF provided responses on 11/10/03. Following NRC acceptance of comments, the final TSTF-422 will be submitted. • The BWROG SE was issued 9/27/02 and the BWROG Topical A version was issued in 2/03. • TSTF-423 was submitted to the NRC on 8/12/03. • TSTF reviewed and provided comments on the BWOG Topical Report. • BWOG Topical Report expected to be submitted to NRC in 11/03. • RITSTF will provide responses to NRC comments on TSTF 423 in advance of the 1/15/04 meeting. • NRC/Industry will meet 1/15/04 on TSTF 422 and TSTF 423.	CEOG - TSTF-422 R0 BWROG - TSTF-423 R0 BWOG - TSTF-431 R0 *(Not created)* WOG - TSTF-432 R0 *(Not created)*

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
2	Missed Surveillances SR 3.0.3	• TSTF-358, R6, has been approved and published for CLIIP adoption.	• Initiative Complete.	TSTF-358 R6
3	Increase Flexibility in Mode Restraints LCO 3.0.4	• TSTF-359, R9, has been approved and published for CLIIP adoption. • The final Implementation Guidance, NEI-03-10, was issued on September 5, 2003.	• Initiative Complete.	TSTF-359 R9
4a	Individual Risk Informed Allowed Outage Times (AOTs)	• Owners Groups (OGs) are pursuing generic Risk Informed AOT extensions through OG-specific Topicals and license amendments.	• Ongoing.	TSTF-373 R2 (CEOG) TSTF-409 R1 (CEOG) TSTF-417 R0 (WOG) TSTF-430 R0 (BWOG) TSTF-439 R1 TSTF-446 R0 (WOG) TSTF-454 R0 (BWROG)

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
4b	Risk Informed AOTs With Configuration Risk Management Program or Maintenance Rule Backstop	TSTF-424 and the Draft Risk Management Guide were provided to the NRC on 1/21/03.	• NRC provided acceptance review comments on the Risk Management Guide on 7/15/03. • NRC provided initial impressions on the STP pilot at the 7/18/03 RITSTF/NRC meeting. • NRC provided acceptance review comments on TSTF-424 on 10/2/03. NRC to provide first round of RAIs in the near future. • RITSTF provided the Interim Report to NRC at the 11/13/03 meeting. • RITSTF will provide the Matrix of the responses to the NRC comments to NRC by 11/21/03. • RITSTF/NRC will meet to discuss the responses and documents 12/16/03.	TSTF-424 R0
5a	Relocate Surveillance Requirements Not Related to Safety	• Deterministic portion of Initiative 5 transferred to TSTF responsibility.	• TSTF reviewing candidate SRs to be relocated. • TSTF will provide a TSTF to the NRC by 6/04.	None assigned

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
5b	Relocate Surveillance Test Intervals to Licensee Control	• RITSTF/BWROG/Pilot Plant will be applying the methodology and interfacing with the NRC on the issues in 2003. • RITSTF to address NRC position that Surveillance Test Intervals must be in the Technical Specifications	• NRC provided a new concern regarding Surveillance Test Intervals at 5/15/03 RITSTF/NRC meeting. NRC stated that OGC was concerned that test intervals (Frequencies) should remain in Technical Specifications under 10 CFR 50.36. It may be acceptable for the Technical Specifications to contain a methodology rather than a fixed interval. • RITSTF to address NRC position that Surveillance Test Intervals must be in the Technical Specifications and NEI to provide a White Paper to NRC. • BWROG has identified two pilot plants and will develop a draft Traveler based on the application of the methodology. BWROG will attach the Guidance Document to the TSTF. • RITSTF provided a draft Technical Guidance Document to the NRC at the 11/13/03 meeting. • NRC will attend the 12/10/03 Limerick IDP meeting. • Three documents will be submitted in 2/04 1. Limerick Pilot as a License Amendment Request. 2. The Methodology Document 3. TSTF 425.	TSTF-425 R0 *(Not created)*

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
6a	Modify LCO 3.0.3 Actions and Timing 1 hour - 24 hours	• On hold.	• On hold for resolution of Initiative 6b and 6c to determine if Initiative 6a is required.	None assigned
6b	Provide Conditions in the LCOs for Those Levels of Degradation Where No Condition Currently Exists to Preclude Entry Into LCO 3.0.3	• NRC drafting Safety Evaluation.	• CEOG provided revised version to address the RAIs to NRC in 7/03. • NRC PSA Branch developed a draft SE and NRC will finalize the SE based on the CE responses. • CEOG to provide a list of ISTS changes and justification to TSTF after NRC completes Safety Evaluation. • TSTF reviewed CEOG Topical Report and recommended changes to support creation of a Traveler. CEOG developed a revision to the Topical to address NRC comments and the TSTF comments. • This revised topical was submitted to NRC 10/3/03. • The revised Topical is under NRC review and NRC plans to issue the draft SE 4/30/04. • TSTF to be submitted after NRC issues SE on Topical.	TSTF-426 R0 *(Not created)*

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
6c	Provide Specific Times in the LCO For Those Conditions That Require Entry Into LCO 3.0.3 Immediately	• NRC drafting Safety Evaluation.	• CEOG provided revised version to address the RAIs to NRC in 7/03. • NRC PSA Branch developed a draft SE and NRC will finalize the SE based on the CE responses. • CEOG to provide a list of ISTS changes and justification to TSTF after NRC completes Safety Evaluation. • TSTF reviewed CEOG Topical Report and recommended changes to support creation of a Traveler. CEOG developed a revision to the Topical to address NRC comments and the TSTF comments. • This revised topical was submitted to NRC 10/3/03. • The revised Topical is under NRC review and NRC plans to issue the draft SE 4/30/04. • TSTF to be submitted after NRC issues SE on Topical.	TSTF-426 R0 *(Not created)*

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
7a	Impact of Non Technical Specification Design Features on Operability Requirements - Barriers	• NRC provided comments on TSTF-427 on 6/26. • TSTF and SNUG have drafted a revision to TSTF-372 that addresses NRC and Industry concerns. Scheduled to submit in early November.	• Draft Revision 4 of TSTF-372 created on 2/24/03. NRC reviewed and agreed it addressed their comments. Draft being reviewed by TSTF, RITSTF, and Snubbers Users Group (SNUG). • TSTF, Snubbers User Group (SNUG), and NRC met to discuss TSTF-372 on June 16. SNUG gathered information from their members and worked with the TSTF to draft a revision that addresses the NRC and Industry concerns. Schedule is to submit a revision in early November. • Draft Traveler for Initiative 7a (TSTF-427) drafted for discussion with the RITSTF on 12/18/02 and NRC 12/19/02. • TSTF-427 transmitted to NRC on 3/4/03. • NRC provided comments on TSTF-427 on 6/26/03. RITSTF provided response to NRC comments on 11/7/03. • RITSTF/NRC discussed the RITSTF response at the 11/13/03 meeting. • NRC will provide feedback on RITSTF responses on TSTF 427 by mid January 2004. • NRC will review the RITSTF response and set another meeting to discuss the issues.	TSTF-372 R2 TSTF-427 R0

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
7b	Impact of Non TS Design Features on Operability Requirements – All other SSCs not in Technical Specifications	• A White Paper on the process to address this scope of SSCs is being developed.	• RITSTF will develop a White Paper to outline the process to address this scope of SSCs by 6/04. • RITSTF/TSTF will work with the NRC on a risk informed revision of GL 91-18 and integrate the Initiatives. • An Operability/GL 91-18 Workshop was held on 8/14/03 to discuss these issues. • RITSTF/TSTF will develop a TSTF and submit to NRC by 6/04.	None assigned
8a	Remove or Relocate Systems LCOs That Do Not Meet the 4 Criterion of 10 CFR 50.36 From Technical Specifications	• A White Paper on the application of the 10 CFR 50.36 criteria is being developed.	• NEI 00-04 is being reviewed and will serve as the basis for Criterion 4 application. • RITSTF will develop a White Paper to outline the guidance and methodology based on NEI 00-04 for the application of the four criteria of 10 CFR 50.36 and a list of the systems identified for relocation. RITSTF working on the schedule – current plans are second quarter 2004.	None assigned

RITSTF INITIATIVE STATUS

INITIATIVE	TITLE	INITIATIVE STATUS	NEXT ACTIONS/ SCHEDULE/ RESPONSIBILITY	TSTF NUMBER
8b	Modify 50.36 Rule to Permit Removal or Relocation of Non Risk Significant Systems out of Technical Specifications	• Requires Rulemaking	• RITSTF looking at coordinating Initiative 8b with longer term initiatives given the requirements for rulemaking. • Approach favored by NEI and NRC is making Criterion 4 a "two way door" (e.g., if it doesn't meet Criterion 4, Specification can be relocated even if it meets Criteria 1, 2, or 3).	Not applicable

BWOG - Active in Initiatives 1, 4 and 7

CEOG - Active in Initiatives 1, 4, 5 and 6

BWROG - Active in Initiatives 1, 4, 5 and 8

WOG - Active in Initiatives 1, 4, and 5

INITIAL REVIEW COMMENTS ON
RMTS INITIATIVE 1
TSTF-423, BWR END STATES

With respect to the differences between Topical Report NEDC-32988 and the Safety Evaluation (SE) described in the "Notes" to Table 1, they are consistent with the justification presented in the Topical Report and the staff's SE. The following comment is with respect to Note 11.

1. The SE stipulates the availability of other systems, to maintain defense-in-depth, for some systems (i.e., primary and secondary containment) in order to remain in Mode 3 for repairs. It is appropriate, at a minimum, to mention these other systems in the Bases since approval of the application of this initiative to these TS is based upon the availability of the other systems.

The following two comments are with respect to the proposed revisions of the SE stipulations, discussed in Table 2 of the Traveler.

2. Item 1: The language of the proposed statement should be modified to make it clear that the "primary purpose of entering Mode 3 is for performing short-duration repairs which necessitated exiting the original operating mode." In addition, in discussions on Topical Report NEDC-32988 in preparation for writing the SE, the staff had wanted an LCO time limit, the industry did not. The industry performed a survey to determine how long plants could remain in MODE 3. Based on the survey they performed they were able to convince the staff that in most cases the plants cannot maintain hot shutdown for more than a few days. The staff had asked if this is true what would be the problem with having an LCO. The issue was that one or two plants could remain in hot shutdown for extended times, so the industry wanted the flexibility of not having an LCO, and the staff agreed with the understanding that plants would definitely not remain in the LCO for more than a week. It is appropriate to add, in the Bases of the applicable sections, discussions that state the following:

- The plant must be brought to a Mode (i.e., Mode 3) in which the overall plant risk is minimized.
- The primary purpose of entering Mode 3 should be for accomplishing short-duraton repairs which necessitated exiting the original operating Mode.
- Remaining in the Applicability of the LCO is acceptable for [up to a week because ...].

3. Item 3 - As noted the correct reference is 50.65(a)(4) and not (b)(4).

Risk-Managed Technical Specifications (RMTS) Guidelines

1002965

Interim Report, October 2003

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Risk-Managed Technical Specifications (RMTS) Guidelines, EPRI, Palo Alto, CA: 2003. 1002965.

REPORT SUMMARY

The Electric Power Research Institute (EPRI) has assessed the role of probabilistic risk assessment (PRA) in the regulation of nuclear power plant technical specifications. This report presents nuclear utilities with one example of a technical framework and associated general guidance for implementation of risk-managed technical specifications (RMTS) as a partial replacement of existing conventional plant technical specifications. This report was prepared by EPRI and the Westinghouse Owners Group (WOG) for the nuclear power industry and for potential future reference and application by the Nuclear Energy Institute (NEI). This report is intended for application to both Westinghouse and non-Westinghouse reactor plants. Currently, the nuclear power industry is making much progress in developing risk-informed applications technology. In this environment, this report is published as an interim report with the knowledge that it will likely be updated and upgraded in the future.

Background

Since 1995, the methodology for applying PRAs to risk-informed regulation has been advanced by the publication of many reports. EPRI has published the *PSA Applications Guide* (TR-105396), *Guidelines for Preparing Risk-Based Technical Specifications Change Request Submittals* (TR-105867), *Risk-Informed Integrated Safety Management Specifications (RIISMS) Implementation Guide* (1003116), and *Risk-Informed Configuration-Based Technical Specifications (RICBTS) Implementation Guide* (1007321). NRC published its final policy statement on *Use of Probabilistic Risk Assessment Methods in Nuclear Activities* in the Federal Register, 60, pp 42622-42629, August 16, 1995, and has issued several regulatory guides and Standard Review Plan sections for risk-informed applications of nuclear power plant regulation, specifically, RG 1.177 providing guidance on risk-informed technical specifications programs. Finally, over the past four years, the Nuclear Energy Institute (NEI) has formed the Risk-Informed Technical Specification Task Force (RITSTF) and the Technical Specifications Working Group to address specific issues associated with the process of "risk-informing" plant technical specifications. This report supplements the *PSA Applications Guide* and current RITSTF efforts, and supports EPRI member utilities in effective and efficient development of risk-managed technical specifications (RMTS) implementation programs.

Objectives

- To provide utilities with an approach for developing and implementing nuclear power plant risk-managed technical specifications programs.
- To complement and supplement existing PRA applications methodologies.

Approach

By using available industry and NRC documentation, experienced PRA practitioners developed an approach and methodology for implementing risk-informed technical specifications.

Results

This report presents a recommended approach and technical framework for an effective RMTS program and its implementation following NRC approval. The approach uses an appropriate blend of deterministic and probabilistic methods. The report includes a comprehensive list of references.

EPRI Perspective

This project shows that most nuclear power utilities can apply their current PRAs to develop risk-managed technical specifications for their plants. This interim report presents a general framework, not a specified method, for RMTS program development. The RMTS framework described herein is designed to be consistent with and build upon NEI maintenance rule guidance (Reference 3). The NEI Initiative 4B pilot plant applications will help determine a more specific methodology for RMTS program implementation in the future. Application of the methodology presented herein may be performed in a phased or staged manner, choosing only "high-value" systems initially, thus front-loading the potential benefits of RMTS implementation. It is important to note that a RMTS program should not permit intentional, simultaneous disabling of all trains of any key safety function. EPRI acknowledges and thanks the STP Nuclear Operating Company (STPNOC) for its invaluable technical support of this project.

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Interest Categories

Risk & reliability
Assessment & optimization

Keywords

Probabilistic risk assessment
Risk-informed applications
Probabilistic safety assessment
Technical specifications
Operations
NRC regulations
Licensing

ABSTRACT

In 1995, EPRI published the *PSA Applications Guide* (TR-105396) to help utilities that own and operate nuclear power plants use their PRAs to improve plant safety and resource allocation, and its *Guidelines for Preparing Risk-Based Technical Specifications Change Request Submittals.* In 2000, EPRI published the *Risk-Informed Integrated Safety Management Specification (RIISMS) Implementation Programs* report (1000893), describing the fundamental concepts of a RIISMS program, followed in 2001, by the *Risk-Informed Integrated Safety Management Specifications (RIISMS) Implementation Guide* (1003116), describing the details of RIISMS program implementation. In 2002, EPRI published its *Risk-Informed Configuration-Based Technical Specifications (RICBTS) Implementation Guide* (1007321). The purpose of this report is to supplement the preceding reports to provide specific guidance on how to implement RMTS programs at existing and planned nuclear power plants. This report is organized and presented as follows:

- Section 1 is an overview of the history preceding RMTS programs.
- Section 2 provides the RMTS process description overview.
- Section 3 presents the detailed RMTS guidance approach and methodology.
- Section 4 presents the attributes of a PRA that are required for RMTS implementation.
- Section 5 presents RMTS references.

Appendices provide supporting RMTS program information.

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1 INTRODUCTION

10CFR50.36, "Technical Specifications," requires that the licensee identify Limiting Conditions of Operation (LCOs). These are the minimum functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the Technical Specifications (TS) until the condition can be met. No specific timing requirements were included in the regulation. However, in practice, actions within an LCO are associated with one or more fixed time intervals. Within the context of the plant TS, these time intervals are termed the Allowed Outage Times (AOTs) or Completion Times (CTs). These time intervals were established at the time of plant licensing.

In the 1980s and early 1990s, risk informed changes were approved for a number of plants including Millstone Units 2 and 3; Palo Verde Units 1, 2 and 3; and the South Texas Project. Early activities in integrating risk insights were used in resolving specific industry issues. These activities were sponsored to varying degrees by all Owners' Groups (References 23 and 24). In 1995, the NRC embarked on an initiative to improve regulatory efficiency and enhance public safety by considering risk insights in regulation. The effort resulted in the risk-informed changes to a wide range of regulatory activities including In-Service Testing (IST), In-Service Inspection (ISI), graded Quality Assurance (QA) and the plant TS. The CEOG AOT extension efforts for the Safety Injection Tanks (SITs), Low Pressure Safety Injection (LPSI) System and Emergency Diesel Generators (EDG) (References 25, 26 and 27) became the pilot documents supporting the development of the Regulatory Guide governing risk informed changes to the Plant TS (Reference 13). As experience with risk informed regulation grew, additional Risk-Informed AOT extensions have been granted.

The impetus for a risk-managed Technical Specifications (RMTS) program was the NRC's "Maintenance Rule" (10 CFR 50.65) (Reference 2), specifically, 10 CFR 50.65(a)(4) which states:

"Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to those structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety."

Following industry feedback from a 1998 stakeholders meeting, the NRC recommended that the industry consider an initiative to risk inform the plant TS. In response to that initiative, several public meetings were held to identify the aspects of the TS that are amenable to a risk informed treatment. Currently, the industry is sponsoring eight Risk Management Technical

Specifications (RMTS) initiatives. This report focuses on enabling conditions for the broad based initiative to replace the existing fixed AOTs/CTs with a flexible CT structure controlled within the plant's 10CFR50.65(a)(4) (Reference 2) Maintenance Rule. In a flexible CT structure, most equipment TS conditions allow the component outage time to be determined based on the actual plant state, maintenance needs, and relative risks. Specifically, the general features of the Nuclear Energy Institute (NEI) Technical Specification Task Force (TSTF) Initiative 4B are discussed, and specific risk informed processes that may be needed for successful implementation of this type of TS are identified. Initiative 4B is the industry effort to transition from existing fixed AOTs to flexible CTs. These processes will supplement an existing 10CFR50.65(a)(4) program and could be subsumed within that program. Inclusion of these supplementary processes within a plant's maintenance program will enable better integration, and support of the plant TS.

It is expected that implementation of RMTS will allow utilities to fully utilize risk-informed tools and processes in the management of plant maintenance. These TS enhancements will reduce plant risk by allowing flexibility in prioritization of maintenance activities, improving resource allocation, and avoiding unnecessary plant mode changes. The RMTS under development is specifically directed towards equipment outages and will not change the manner in which plant design parameters are controlled.

This guide essentially refines and supplements Nuclear Energy Institute guidance for implementation of the Maintenance Rule (see Section 11 of Reference 3). Additional key references include EPRI's PSA Applications Guide (Reference 4) and NRC's Regulatory Guide 1.174 (Reference 5).

Maintenance activities must be performed to provide the level of plant equipment reliability necessary for safety, and should be carefully managed to achieve a balance between the benefits and potential impacts on safety, reliability and availability.

The benefits of well managed maintenance conducted during power operations include increased system and unit availability, reduced equipment and system deficiencies that could impact operations, more focused attention on safety due to fewer activities competing for specialized resources, and reduced work scope during outages.

This report is a key part of NEI RITSTF Initiative 4B. Initiative 4B is designed to be consistent with, and provide enhancement to, the guidance provided for maintenance rule risk management described in Reference 3. This section summarizes the enhancements that this initiative brings to prudent safety management.

It is not the intent of the NEI RITSTF initiatives to modify the manner in which the maintenance rule requirements are met by various utilities. However, it is the intent of this report to provide the guidance for integrating risk managed technical specifications with the maintenance rule process. While the fundamental process to be used for the flexible TS is not different from the maintenance rule process, the proposed risk assessment process has an increased quantitative focus and requires a more formal mechanism for dispositioning maintenance decisions. These

features balance the flexibility in performing maintenance within a structural risk informed framework so as to adequately control the risk impact of plant maintenance decisions.

The RMTS process discussed in this report may be used within the current structure of the maintenance rule. Specifically, this report describes a proposed integration of the present 10CFR50.65(a)(4) evaluation process with selected supplementary processes to create an enhanced process that will support the implementation of Flexible CTs within the plant TS.

The RMTS features/processes to be integrated into a risk informed application of the (a)(4) risk management process include:

1. Identification of, and timely response to, emergent (unscheduled) risk-significant conditions.
2. Assessment of potential common cause failure risk associated with emergent failures.
3. Implementation of a formal Risk Informed Decision Process for plant shutdown/mode change.
4. Process for consideration of external events (Reference 20) and risk significant fire events.

The first feature is added to ensure that a delay in the risk assessment (up to the time of the front-stop) will not result in the emergence of a high-risk plant configuration. The second and third enhancements are defined to support the assessment that the "configuration is acceptable for continued operation" beyond the front-stop. The addition of a formal external events (Reference 20) risk assessment process is intended to ensure proper disposition of the risk informed decision.

It should be noted that many existing maintenance rule programs include one or more of the above features already. However, it is expected that some accommodations and enhancements may be required to existing maintenance rule applications for RMTS implementation. The addition of these processes to the existing 10CFR50.65(a)(4) (where they do not already exist) risk assessment process will facilitate the transition to flexible CTs.

The integrated process is intended to provide a comprehensive risk informed mechanism for expeditious identification of risk significant plant conditions. This will include the implementation of appropriate risk informed maintenance actions, and may include the action to shutdown the plant. In practice, this program is transparent for all 10CFR50.65(a)(4) maintenance planning conditions. That is, the program retains the current 10CFR50.65(a)(4) practice, which prohibits the plant from voluntarily entering high-risk conditions without proper evaluation of the concurrent risks and implementation of appropriate management actions. Enhancements are associated with emergent (unscheduled) maintenance states and are recommended to ensure that high-risk conditions associated with multiple component outages are identified early and that a risk-informed process exists to affect a shutdown, as appropriate.

2 PROCESS DESCRIPTION

This document has been developed to provide the commercial nuclear power industry guidance on risk management issues associated with implementation of risk-management Technical Specifications (RMTS) programs at their facilities. Specifically, this guide is designed to support the implementation of a risk-informed approach to the management of equipment "allowed outage time" (AOT) or maintenance "completion time" (CT) related to safety functions addressed by plant technical specifications. Henceforth, in this document, we will refer to AOT and/or CT simply as CT. See Appendix A of this guide for a glossary of key terms applicable to RMTS program development and implementation.

The RMTS process presented in this report integrates the appropriate regulatory guidance. The overall maintenance risk will be assessed via processes consistent with 10CFR50.65, and its attendant Regulatory Guide (RG) 1.182. Risk informed front-stop times will be established based on single SSC outage guidelines of RG 1.177 or using the traditional non risk-informed standard Tech. Specs.. In addition, the overall use of the RMTS process will be periodically assessed to demonstrate compliance with RG 1.174 guidance for small risk impact plant modifications (i.e., yearly change < 10^{-5} per year).

Existing conventional technical specifications for nuclear power plants specify required maximum CT values for specific plant equipment related to the maintenance of key plant safety functions. Under the proposed RMTS concept, these CT values would be maintained and referred to as "front-stop" CT values. However, operation beyond the front-stop would be allowable provided the risk of continued operation can be shown to remain within established safety limits. The process for allowing continued operation will involve performance of risk assessments and definition of risk-informed CT (RICT) targets and limits. The RICT is the time from the initiation of a maintenance configuration until a risk threshold or limit (described in Section 3) is reached. The RICT will have an ultimate maximum CT limit (currently established at 30 days), referred to as the "back-stop" CT. The front-stop CT values may be either those that have historically been established via conventional deterministic engineering methods and judgment or those more recently justified via RG 1.177. The back-stop CT limit of 30 days is judged to be a prudently conservative administrative limit for configuration risk management, as compared with the 10CFR50.59 design change criteria limit of 90 days. It is anticipated that application of RICTs for individual maintenance configurations would realistically rarely exceed approximately two weeks. The front-stop CT, RICT, and back-stop CT taken in conjunction can be thought of as a type of "defense-in-depth" approach to maintenance configuration and associated technical specification risk management. The proposed approach builds upon the recognized need that the maintenance of equipment in nuclear power industry could benefit from the application of current "state-of-the-art" risk management methods, tools, and techniques.

In a RMTS program, the structure of the risk-informed technical specifications (RITS) will be similar to familiar traditional TS with the exception that actions will be provided to allow continued plant operation beyond the TS "front-stop". Thus, if a need arises, plant operators would have an option of exceeding that CT provided a risk assessment confirms the risk is reasonably expected to remain within established safety limits. Guidance for continuing maintenance beyond the CT must be consistent with the Maintenance Rule Guidance, and the risk associated with this continued maintenance must be tracked. Risk assessments should be performed in accordance with the plant's Maintenance Rule program and supported by a plant's PRA and other risk management tools (e.g., plant safety monitor or risk monitor software, risk matrices, lists of pre-analyzed maintenance configurations, PRA sensitivity studies, etc.) for specified hazards and operational plant states. These tools are typically applied in 10CFR50.65(a)(4) and 10CFR50.59 assessments and evaluations. The term "maintenance configuration" is defined in Appendix A and is simply the consolidated state of all plant equipment along with their states of functionality, i.e., either functional or non-functional (definition in Appendix A), and applies to both preventive and corrective maintenance.

Risk-managed LCOs will be entered when the associated TS components are declared inoperable. The assignment of inoperability will follow current TS guidance. Once the LCO is entered, the functional impact (related to SSC availability to support its/their applicable safety function(s)) of the inoperability will be considered within the scope of the risk assessment. For example, HPSI inoperability may vary in risk significance, dependent on the degree of residual capability (capability to support required safety functions) of the system.

A target and maximum RICT would be calculated before the front-stop CT limit is reached. The target RICT would be used to track the success of the maintenance activity consistent with normal work controls. Consistent with NEI maintenance rule guidance (Reference 3), a target RMTS configuration risk would be a configuration incremental Core Damage Probability (ICDP) of 10^{-6} (as measured from the time the associated plant equipment goes out of service). For emergent conditions (or for forced, unscheduled extension of planned maintenance) a maximum RICT equivalent to an ICDP of 10^{-5} is identified. Under no circumstances is the RICT to exceed the ultimate back-stop CT limit of 30 days (as measured from the time of entry into the associated TS LCO front-stop CT). The use of administrative maintenance target risk values at levels significantly below the RICT will ensure adequate margin to unanticipated concurrent failures.

In an RMTS program, a RICT would only be calculated when the plant enters a TS LCO associated with a specific plant maintenance configuration (see definition in Appendix A), and it is determined that completion of maintenance allowing exiting that LCO would not be practicable within the front-stop CT. If, during application of a specified RICT, the plant transitioned to a different maintenance configuration (e.g., due to emergent conditions), then that RICT would be required to be recalculated and revised within a specified time period (24 hours, for example) after the change in configuration. It is important to note that this 24-hour re-assessment period is simply an example applied in this report. Case-specific re-assessment periods applied within a plant-specific RMTS program will need to be consistent with the application of associated front-stop CT requirements. This revised RICT would be effective from the time of implementation of the original RICT for the original maintenance configuration,

and the associated maintenance "time-clock" would not be re-set to zero at the time of the modified configuration. In the RMTS framework, a RICT can be revised, occasionally many times, but not exited (or re-set to the remaining licensing period duration) until the plant satisfactorily exits all TS LCOs where the associated front-stop CT has been exceeded. If a revised RICT were found to exceed a RMTS threshold, the plant would re-evaluate the impact and enter a plant shutdown decision process. If the revised RICT exceeds the upper-level RMTS threshold based on specified ICDP/ILERP limits (see Section 3, specifically Table 3-2), the plant would be required to take the actions required for "ACTION NOT MET" for the affected Technical Specifications. Note that, during the time period following the front-stop CT but before the expiration of the applicable RICT, plant actions will escalate to be commensurate with the projected risk during the maintenance configuration period, consistent with the current maintenance rule guidance (Reference 3).

In a RMTS program, the conventional technical specification definition of equipment "operability" (see Appendix A) applies, just as it does under current existing plant technical specifications. Thus, equipment "operability" is applied by plant operating staffs to enter or exit TS LCOs. However, the issue of equipment "functionality" (see Appendix A) is broader and relates more directly to the equipment's availability to support its intended risk mitigation function. Equipment functionality will generally be considered in a RMTS program when assessing risk for RICT calculation.

3 GUIDANCE

This section describes an approach to support RMTS by estimating the overall risk of potential plant configurations and by providing information to plant personnel to that they can take appropriate actions to control it.

10CFR50.65(a)(4) requires that a risk assessment be performed prior to performing maintenance. The scope of the RMTS generally includes, at a minimum, SSCs modeled in a Level 1 PRA and other SSCs that have been determined to be of high safety significance via processes outlined in Reference 3.

The (a)(4) process uses PRA methods and risk insights in establishing and planning maintenance activities. The RMTS program recommended herein meets (a)(4) requirements by using existing (a)(4) guidance in many areas and by implementing a more rigorous application in the remaining areas. The following guidance would replace the existing (a)(4) guidance for plants implementing RMTS.

3.1 Assessment Process, Control, and Responsibilities

10CFR50.65 paragraph (a)(4) states that "before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities." Risk assessments are not limited to temporarily inoperable equipment but can include equipment troubleshooting, hazard barrier removal, erection of scaffolding, lifting electrical leads and installing electrical jumpers. The scope of the assessment may be limited to SSCs that a risk-informed evaluation process has shown to be significant to public health and safety. Furthermore, the (a)(4) plant maintenance evaluation is required for all plant operating modes and considers the impact of external events (such as fire, seismic, flooding, high wind, and other initiating events defined as external events in Reference 20). Additional details of this process are contained in NUMARC-93-01 (Reference 3).

The RMTS process shall:

1. Be documented in plant procedures delineating appropriate responsibilities for (a)(4) related actions.

2. Include procedures for performing a risk assessment when the maintenance items are outside the scope of the quantitative risk assessment tool.

3. Define a process so that when the plant configuration is outside the scope of the RMTS quantitative calculation tool (e.g., risk or safety monitor software, a risk matrix, a list of pre-calculated risk levels for specified maintenance configurations, etc.), or the tool is otherwise unavailable, qualitative methods may be used to assess risk and define appropriate actions to manage the risk increase.

4. Include guidance for using risk insights to manage overall plant risk.

In performing the RMTS assessment, the decision-making process will, where appropriate, include consideration of transition risks associated with mode changes. Consideration of mode transition risk is appropriate when a mode transition is actually involved in the implementation of maintenance, and when the calculation of the maintenance configuration risk would not be bounded (as an upper bound) by a calculation of the steady-state at-power risk for the configuration of interest.

Implementation of the RMTS risk assessment process requires integration into the plant-wide work control process. The process then requires identification of the current plant maintenance configuration and performance of a risk assessment applicable to that configuration. Appropriate actions to manage the risk impacts shall then be determined and implemented.

The remainder of this report assumes that the plant is fully compliant with 10CFR50.65(a)(4). It is further assumed that the plant risk assessments integrate PRA results and PRA-derived risk insights into the process. The supplementary processes discussed in this report are intended to enhance the existing (a)(4) process in order to allow it to accommodate a greater plant control function. The primary intent of these processes is to ensure that selected desirable attributes of the current TS are pragmatically retained in the RMTS structure in a Risk Informed Format. These attributes include:

1. Current (conventional) TS Structure
2. Reliance on defined time interval limits (i.e. front-stop CT)
3. Reference to defined actions in an LCO

The RMTS is intended to replace the fixed CT and the prescriptive actions of the current TS with an action statement to conduct a risk informed assessment. The structure of the proposed RMTS is illustrated in Table 3-1. Note that the proposed TS references three time intervals: the front-stop CT, the 30-day (or "back-stop") CT, and the acceptable risk informed time interval (the RICT) calculated in accordance with the RMTS thresholds (see Table 3-2). The front-stop is the plant's TS CT as justified via design basis considerations or TS CT as modified via an approved RG 1.177 analysis. The 30-day completion time is provided to ensure the plant design basis is retained (that is, no permanent plant changes are made associated with this TS). The 30-day interval is not risk informed, but rather represents a deterministic limit. The level of acceptable risk beyond the front-stop is established via a risk-informed application of maintenance rule guidance (Reference 3) as follows:

1. Prior to exceeding the front-stop CT, the plant must perform a configuration risk assessment to confirm that the risk of continued operation in the existing configuration is

acceptable. A quantitative/qualitative risk assessment will provide the basis for continued plant operation. The assessment must consider the impact of common cause failures, external events (Reference 20) and flooding. In addition, the assessment may credit compensatory actions established during the period being evaluated.

2. Depending on the outcome of this assessment and assessment of alternative actions, risk management actions will be defined and the plant will either continue operation beyond the front-stop CT or take other action in accordance with TS. The timing of the plant shutdown will reflect plant cumulative risks, the likelihood of repair and transition and shutdown risk considerations.

For emergent (unscheduled) conditions, the plant staff is expected to provide an expeditious assessment of the plant risk. Typically, such an assessment should be performed within 24 hours of an emergent condition. Quantitative risk assessments will be performed with an appropriately contemporaneous (as-built, as-operated) plant risk model, and PRA results should be based on PRAs with Level 1 and 2 attributes (adequate for the assessment of maintenance configuration impacts on CDF and LERF) compatible with the associated risk informed application. Fire, seismic and/or flood risks must also be considered when establishing the duration of a proposed extension.

Conceptually the implementation of the flexible CT is simple. For all entries into the RMTS, the licensee will:

1. Perform risk assessment and management in accordance with the Maintenance Rule (a)(4),

2. Prior to the expiration of the TS front-stop CT, a risk assessment of the maintenance configuration resulting from the inoperable equipment will be performed by using the RMTS guidance to determine the feasibility of continued power operation beyond the front-stop,

3. Based on the results of the risk assessment, the plant staff will take actions to manage risk by potentially expediting repairs, or by implementing contingency actions, including initiating a plant shutdown, and

4. Once the extended CT is entered, the RMTS risk assessment will be re-performed in accordance with the RMTS program when emergent conditions change the evaluated maintenance configuration. In the event of an emergent condition, the assessment should be performed within 24 hours following any plant maintenance configuration changes. The risk management program will be based on a risk informed application of 10CFR50.65(a)(4).

The risk assessment process will focus on the entire maintenance evolution and will utilize the quantitative action thresholds of Section 11.3.7.2 of Reference 3. In addition, risk assessments will be performed to assess the incremental risk of the inoperable equipment associated with maintenance configuration addressed by the extended CT (RICT). These latter results will be

tracked, trended, and periodically reviewed to ensure that the cumulative risk of the flexible TS is small (per RG 1.174). Furthermore, this process will reduce the potential for performing higher risk maintenance beyond the front-stop. For conditions where risk consideration alone would result in a very long RICT, restoration of low risk, design basis configurations/equipment will be ensured by the back-stop CT.

The process for conducting and using the result of the risk assessment in plant decision-making will be documented in an approved plant procedure, and each assessment supporting implementation will be documented. An example general process flowchart for RMTS risk assessment and implementation is presented in Figure 3-1. The procedures should specify the plant functional organizations and personnel, including operations, engineering, and risk assessment (PRA) personnel, responsible for each step of the procedures. The procedures should also clearly specify the process for conducting, reviewing, and approving the assessment. In all cases where a RICT assessment cannot be performed (e.g., when the configuration risk cannot be adequately addressed via the CRMP and PRA), the normal front-stop CT(s) will be applied.

For plants implementing an RMTS program, it is generally a good practice to develop and maintain a "pre-analyzed" list of maintenance configurations with associated RICT values. This list does not necessarily need to address all SSCs affected by TS LCOs, but should address reasonable combinations of disabled safety function equipment trains and instrument channels.

TABLE 3-1
GENERIC RISK-INFORMED CTs WITH A BACK-STOP: EXAMPLE FORMAT

Actions Condition	Required Action	Completion Time
B. One [HPSI] subsystem inoperable.	B.1 Restore SI subsystem to OPERABLE status.	72 hours
	<u>OR</u>	
	B.2.1 Determine that the completion time extension beyond 72 hours is acceptable in accordance with established RMTS thresholds.	72 hours
	<u>AND</u>	
	B.2.2 Verify completion time extension beyond 72 hours remains acceptable.	In accordance with the RMTS Program (i.e., within 24 hours of a subsequent configuration change)
	<u>AND</u>	
	B.2.3 Restore subsystems SI to OPERABLE status.	30 days or acceptable completion time , whichever is less.

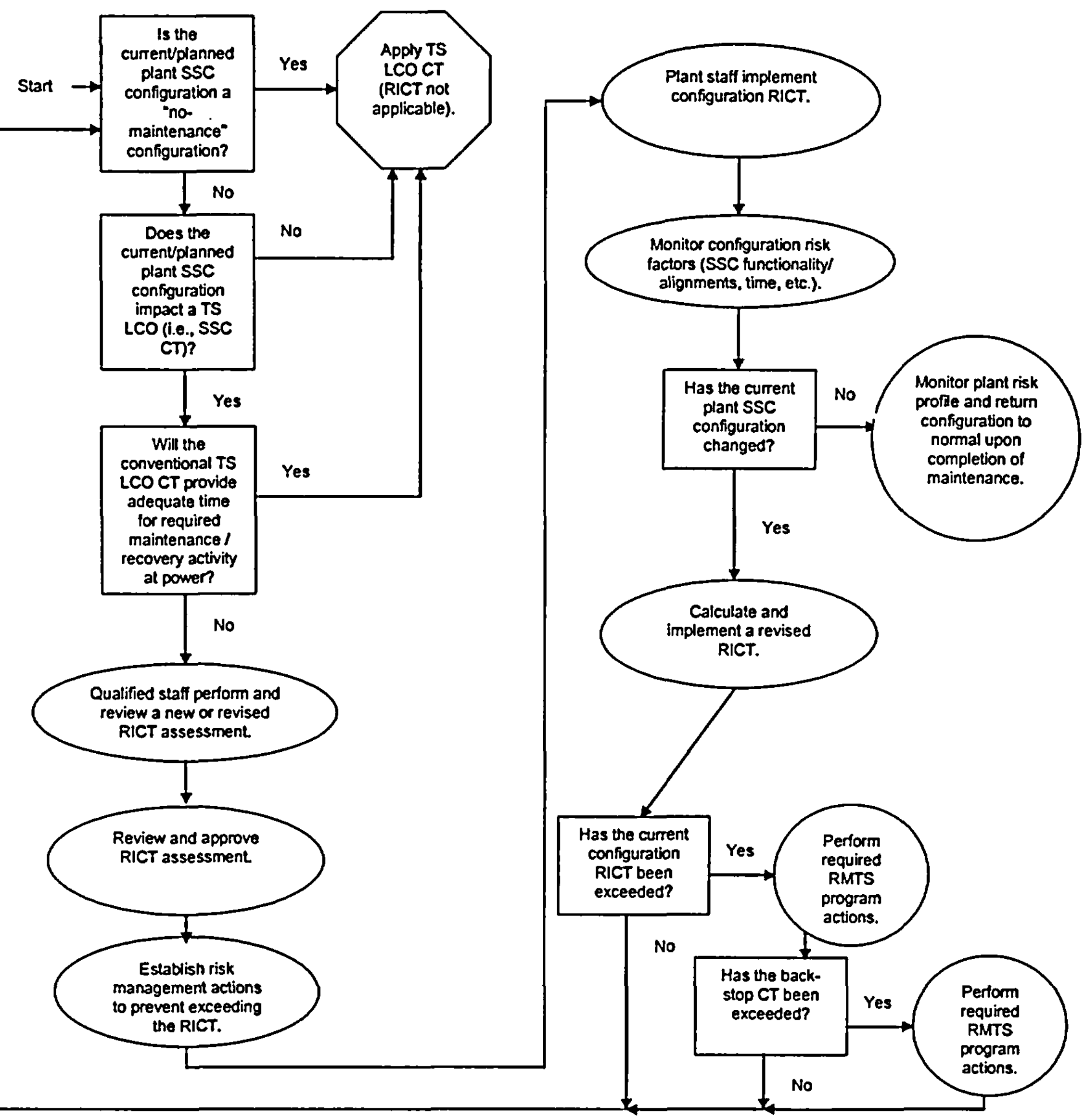

Figure 3-1
Example Process Flowchart for RMTS RICT Assessment and Implementation

3.2 General Guidance for the Assessment

1. Power Operating Conditions are defined as plant modes other than cold shutdown, refueling, or defueled. Section 3.3 describes the scope of SSCs subject to the assessment during power operations.

2. The risk assessment method may use quantitative approaches supported by qualitative approaches.

3. The quantitative assessment should be based on the plant Maintenance Rule risk assessment program supported by the plant PRA. In specific instances, bounding assessments may be appropriate (i.e., in cases where a simplified bounding risk assessment is convenient and can show that a lower bound RICT calculated via an upper bound configuration risk yields ample time for maintenance implementation).

4. The assessment must consider:

   - Technical Specifications requirements
   - Availability of other equipment to perform the safety function(s) served by the out-of-service SSC
   - Potential for common cause failure of redundant equipment
   - The anticipated duration of the out-of-service or testing condition
   - The likelihood of an initiating event or accident (including both internal and external events as defined in References 19 and 20) that would require performing the affected safety function
   - The likelihood that the plant maintenance configuration will significantly increase the frequency of a risk-significant initiating event (References 19 and 20) (as determined by each licensee, consistent with its obligation to manage maintenance-related risk)
   - Component and system dependencies that are affected
   - Significant performance issues for the in-service redundant SSCs
   - Significant industry experience related to the maintenance configuration of interest (Note that updating input information from industry experience should be consistent with Maintenance Rule (a)(4) assessment and PRA updates for the plant.)
   - Compensatory actions taken to mitigate the risks, e.g. alignment of cross-ties with other units, installation of temporary systems.

5. In determining the risk impact of the plant configuration, the assessment may also consider the following factors:

   - the risk impact of the configuration during shutdown versus performing the maintenance at power.
   - the impact of plant mode transition risk (the cumulative risk incurred during one or more associated plant mode changes) if the equipment outage would require one or more associated mode changes that would otherwise be unnecessary.

6. Assessments may be predetermined or performed on an as-needed basis.

7. The degree of depth and rigor used in managing risk should be commensurate with the complexity of the planned configuration and the level of expected risk.

8. Maintenance may involve altering the facility or procedures for the duration of the maintenance activity. Examples of such alterations include jumpering terminals, lifting leads, placing temporary lead shielding on pipes and equipment, removing barriers, and using temporary blocks, bypasses, scaffolding and supports. The assessment should include consideration of the impact of these alterations on plant safety functions qualitatively or quantitatively depending on the significance of the alteration.

9. For surveillance testing or situations where the maintenance activity has been planned in such a manner to allow for prompt restoration of SSC functions, the assessment may take into account the likelihood and restoration time of out-of-service SSCs being promptly restored to service in response to emergent conditions. In this context, the terms "prompt" and "promptly" mean that the restoration of SSC function occurs prior to its associated demand for risk mitigation, given the occurrence of a predicted emergent condition or event sequence that affects plant safety (and risk).

10. Emergent conditions(or forced, unscheduled extension of planned maintenance) may require action prior to completing the assessment, or could change the conditions of a previously performed assessment. Examples include plant configuration or mode changes, additional SSCs out of service due to failures, or significant changes in external conditions e.g. weather, offsite power availability, etc.. The following guidance, consistent with Reference 3 guidance, applies to such situations:

    - The safety assessment should be performed (or re-evaluated) to address the changed plant conditions on a reasonable schedule commensurate with the safety significance of the condition. Procedural guidance must be provided in the plant's RMTS program request submittal to specify the appropriate completion time for reassessing the risk. Based on the results of the assessment, ongoing or planned maintenance activities may need to be suspended, modified or rescheduled, and SSCs may need to be returned to service.
    - Performance (or re-evaluation) of the assessment should not interfere with, or delay, the operator and/or maintenance crew from taking timely actions to restore the equipment to service or take compensatory actions.
    - If the plant configuration is restored prior to the required re-evaluation risk assessment (e.g., within 24 hours), the assessment need not be performed for purposes of supporting that maintenance activity. However, an evaluation should be considered in the plant's administrative program for controlling cumulative or aggregate risk (see Section 3.5.2).

3.3 Scope of RMTS and RMTS Assessment

The NRC Maintenance Rule requirements for plant maintenance configuration risk assessment are stated in 10 CFR 50.65(a)(4). 10 CFR 50.65(a)(4), states "The scope of the Systems,

Structures and Components (SSCs) to be addressed by the assessment may be limited to those SSCs that a risk-informed evaluation process has shown to be significant to public health and safety." Thus, the scope of SSCs subject to the RMTS assessment provision may not include all SSCs that meet sections (b)(1) and (b)(2) maintenance rule scoping criteria.

From a practical standpoint, a RMTS program effectively defines its scope of equipment maintenance configurations (see definition of "maintenance configuration" in Appendix A) to be those associated with SSCs that are included within the scope of current technical specifications LCOs, and therefore, have front-stop CT requirements, but excludes fundamental technical specifications safety limits and limiting safety system settings (e.g., reactivity control, power distribution parameters, etc.).

The PRA provides an appropriate primary mechanism to define the RICT assessment scope, as the PRA scope considers dependencies and support systems, and, through definition of top events, cut sets, and recovery actions, includes those SSCs that could, in combination with other SSCs, result in significant risk impacts. Thus, the risk informed assessment scope may be limited to the following scope of SSCs:

1. Those SSCs included in the scope of the plant's Level 1 (or Level 2 if available), internal (and, if available, external) events PRA, and;

2. SSCs in addition to the above that have been determined to have high safety significance through the process described in Section 9.3 of NUMARC 93-01 (Reference 3).

The PRA used for the RMTS scope and risk assessment should have the following characteristics:

- The PRA should meet adequacy criteria such as industry standards for risk-management technical specification applications (see References 19, 20, and 21). Specifically, the PRA should meet ASME PRA standard (Reference 19) requirements for effective risk evaluation of all SSCs within the scope of the RMTS program. The NRC has recognized Reference 19 standards in DG-1122 (Reference 21).
- Some SSCs within the plant PRA scope may be determined to have low safety significance regardless of plant configuration. These SSCs need not be included in the scope of the risk assessments. Pre-existing analyses and/or expert panels may be used to facilitate these determinations.
- A process for assessing important large early release frequency (LERF) considerations (containment performance, release category frequencies, etc.) should be applied in addition to an acceptable Level 1 PRA.

3.4 Assessment Methods for Power Operating Conditions

Removal of a single SSC from service for longer than its front-stop CT, or simultaneous removal from service of multiple SSCs for longer than the most limiting front-stop CT, requires an assessment using blended (quantitative supported by qualitative) methods consistent with

Reference 3 guidance. Sections 3.4.1 and 3.4.2 provide guidance regarding quantitative and qualitative considerations, respectively.

3.4.1 Quantitative Considerations

1. The assessment process shall be performed via a tool or method that considers quantitative insights from the PRA. Acceptable tools include the PRA model, safety/risk monitor, risk matrix, or pre-analyzed list derived from the PRA insights. To properly support the assessment, the PRA must have certain attributes, and it must reasonably reflect the plant configuration. Section 5 provides information on PRA attributes. Section 3.5.2 provides guidance on various approaches for using the output of a quantitative assessment to manage risk.

2. If the PRA does not directly model the SSC to be removed from service (e.g., the SSC is part of the RPS system, diesel generator, etc. which has been modeled as a "single component"), the assessment should consider the impact of the out of service SSC on the safety function of the modeled component. SSCs are considered to support the safety function if the SSC is significant to the success path for function of the train or system (e.g., primary pump, or valve in primary flow path). However, if the SSC removed from service does not contribute to the unavailability of the associated train or system safety function (e.g., indicator light, alarm, drain valve), the SSC would not be considered to support the safety function.

3.4.2 Qualitative Considerations

1. The quantitative assessment should be supplemented by a qualitative evaluation. For example, the impact of the maintenance activity upon key safety functions, may be addressed as follows:

   - Identify key safety functions affected by the SSC planned for removal from service
   - Consider the degree to which removing the SSC from service will impact the key safety functions
   - Consider degree of redundancy, duration of out-of-service condition, and appropriate compensatory measures, contingencies, or protective actions that could be taken, if appropriate, for the activity under consideration.

2. For power operation, key plant safety functions are those that ensure the integrity of the reactor coolant pressure boundary, ensure the capability to shut down and maintain the reactor in a safe shutdown condition (see definition in Appendix A), and ensure the capability to prevent or mitigate the consequences of accidents that could result in potentially significant offsite exposures.

3. The key safety functions are achieved by using systems or combinations of systems. The configuration assessment should consider whether the maintenance activity would:

   - Affect the likelihood of a PRA initiating event (References 19 and 20)
   - Involve a significant potential to cause a scram or safety system actuation

- Result in significant complications to recovery efforts.

4. Depending on the level of anticipated configuration risk, risk impacts of equipment outages may be determined via approximate or bounding analyses.

5. Qualitative considerations may also be necessary to address external events (Reference 20), and SSCs not in the scope of the Level 1, internal events PRA (e.g., included in the assessment scope because of expert panel considerations).

6. The assessment may need to consider actions that could affect the ability of the containment to perform its function as a fission product barrier. With regard to containment performance, the assessment should consider:

   - Whether new containment bypass conditions are created, or the probability of containment bypass conditions is increased
   - Whether new containment penetration failures that can lead to loss of containment isolation are created
   - Whether redundant or diverse containment safeguards should be available, if maintenance is performed on SSCs of the containment systems (or SSCs upon which containment functions are dependent).

7. External event (Reference 20) considerations involve the potential impacts of weather or other external conditions relative to the proposed maintenance evolution. In the assessment, weather, external flooding, and other external impacts need to be considered if such conditions are imminent or have a high probability of occurring during the planned out-of-service duration. An example where these considerations are appropriate would be the long-term removal of exterior doors, hazard barriers, or floor plugs.

8. Flooding considerations (from internal or external sources) should be addressed if pertinent. The assessment should consider the potential for maintenance activities to cause internal flood hazards, and for maintenance activities that expose SSCs to flood hazards that degrade their capability to perform key safety functions.

3.5 Managing Risk

Risk management uses the risk assessment in plant decision-making to control the risk impact. This process requires careful planning, scheduling, coordinating, monitoring, and adjusting of maintenance activities.

The objective of risk management is to control the temporary and aggregate risk increases from maintenance activities. This control is accomplished by using target and maximum limit RICT values (calculated based on the risk thresholds presented in Table 3-2) to plan and schedule maintenance such that the risk increases are limited, and to take additional actions beyond routine work controls to address situations where the temporary risk increase is above specified RMTS thresholds (see Table 3-2). These thresholds may be set on the basis of qualitative considerations (e.g., remaining mitigation capability), quantitative considerations (e.g., temporary increase in core damage frequency), or blended approaches using both qualitative and quantitative insights.

Management of risk also considers aggregate risk impacts. (Aggregate risk is the collective risk impact over time. Aggregate risk is also known as cumulative risk.) Aggregate risk is controlled to a degree through maintenance rule guidance (Reference 3) requirements to establish and meet SSC performance criteria. These requirements include considering the risk significance of SSCs in establishing performance goals. Plants that implement RMTS should develop measures to assess the aggregate risk relative to the average risk. This assessment could be accomplished through a periodic assessment of previous out-of-service conditions. Such an assessment may involve quantitatively estimating cumulative risks or may involve qualitatively assessing the risk management approach employed.

The PRA provides valuable insights for risk management, because it relates events and systems. Risk management can often be effectively supported by using qualitative insights from the PRA, rather than sole reliance on quantitative information. Removing equipment from service may alter the significance of various risk contributors from those identified via a typical PRA designed to calculate average annual risk. Specific configurations can result in increased importance of certain initiating events (References 19 and 20), or of systems and equipment used to mitigate accidents. Evaluating specific configurations can identify "low order" cut sets or sequences that may not be important in the "annual average" risk analysis but become important for a specific configuration. These considerations are very important to risk management within a RMTS program.

The most fundamental risk management action is planning maintenance activities with the insights provided by the assessment. In conjunction with scheduling the sequence of activities, compensatory risk management actions may be taken that reduce the temporary risk increase. Since many of the risk management actions involve non-quantifiable factors, the risk reduction would not necessarily be quantified. The following sections discuss the establishment of thresholds for the use of risk management actions.

3.5.1 Establishing action thresholds based on qualitative considerations

In accordance with Reference 3, risk management action thresholds (i.e., plant conditions and associated configuration risk levels determining when risk management actions are required) may be established qualitatively by considering the performance of key safety functions, or the remaining mitigation capability, given the out-of-service SSCs. Qualitative methods to establish risk management actions would generally be necessary to address SSCs not modeled in the PRA. This approach typically involves consideration of the following factors in the assessment:

- Duration of out-of-service condition, since longer duration results in increased exposure time to initiating events
- The type and frequency of initiating events that are mitigated by the out-of-service SSC, considering the sequences for which the SSC would normally serve a safety function
- The impact of the plant configuration on the initiating event frequencies
- The number of remaining success paths (redundant systems, trains, operator actions, recovery actions) available to mitigate the initiating events
- The likelihood of proper function of the remaining success paths

The above factors can be used as the basis for establishing a matrix or list of configurations and associated risk management actions.

3.5.2 Establishing action thresholds based on quantitative considerations

3.5.2.1 Quantitative risk action thresholds

The thresholds for risk management actions may be established quantitatively by considering the magnitude of the instantaneous core damage frequency (CDF_{inst}), incremental core damage frequency (ICDF), and the incremental large early release frequency (ILERF) for the maintenance configuration of interest. Plants should consider factors of duration in setting the risk management acceptance guidance. Duration may be either a particular out-of-service condition or a specific defined work interval (e.g. shift, day, week, etc). The product of the configuration ICDF or ILERF and the effective duration of the associated configuration is expressed as a probability configuration incremental core damage probability ($ICDP_{config}$) or configuration incremental large early release probability ($ILERP_{config}$) respectively.

Guidance for evaluating temporary risk increases by considering configuration-specific CDF_{inst}, as well as $ICDP_{config}$ and $ILERP_{config}$ is provided in NUMARC 93-01, Revision 3 (Reference 3). When combined with the other elements of maintenance rule guidance, and other quantitative or qualitative measures, this guidance is acceptable for RICT implementation:

1. Maintenance configurations with a configuration-specific CDF_{inst} in excess of 10^{-3} per year should be carefully considered before voluntarily entering such conditions. If such conditions are entered, it should be for very short periods of time and only with a clear detailed understanding of which events dominate the risk level.

2. Quantitative risk acceptance guidelines using $ICDP_{config}$ and $ILERP_{config}$ for a specific maintenance configuration are presented in Table 3-2. The quantitative risk acceptance guidelines presented in Table 3-2 are consistent with NEI Maintenance Rule (a)(4) guidance (Reference 3). These risk acceptance guidelines should be considered with respect to establishing risk management actions.

TABLE 3-2
RMTS QUANTIATIVE RISK ACCEPTANCE GUIDELINES

Criterion		Risk Management Guidance
$CDF_{inst} > 10^{-3}$/yr		- Careful consideration before entering the configuration
ICDP	ILERP	
$>10^{-5}$	$>10^{-6}$	– Configuration should not normally be entered voluntarily
10^{-6} - 10^{-5}	10^{-7} - 10^{-6}	– Assess non-quantifiable factors – Establish risk management actions
$< 10^{-6}$	$<10^{-7}$	– Normal work controls

In a RMTS program the 10^{-6} and 10^{-7} thresholds for ICDP and ILERP, respectively, are referred to as "target" or "lower-level" RMTS thresholds, while the 10^{-5} and 10^{-6} thresholds for ICDP and ILERP, respectively, are referred to as "maximum limit" or "upper-level" RMTS thresholds.

This guide provides a risk management scheme based on incremental risk metrics as supported by application of Reference 3, Reference 4, 10 CFR 50.65(a)(4), and related maintenance configuration risk management policies currently in effect. Individual plants may choose to propose application of a similar risk management scheme based on absolute risk metrics versus incremental risk metrics.

Using this framework for risk management, the plant staff can calculate target and maximum risk-informed allowed outage times. For planned maintenance, target outage times should be established at low risk levels and should be accompanied by normal work controls. The process to manage the risk assesses the rate of accumulation of risk in plant configurations and determines acceptability of continued plant operation (beyond the front-stop) based on risk assessment, alternative actions and the impact of compensatory actions.

The application of a configuration-specific RICT is strictly a "configuration-based" risk management activity. That is, a specified RICT is directly associated with an "off-normal" plant SSC configuration that is considered temporary. The RICT is to be used in concert with "aggregate" or longer-term "cumulative" risk management policies, based on qualitative and quantitative criteria described in References 4 and 5. One example of an acceptable plant administrative policy designed to effectively implement cumulative risk management is described in Reference 1. Application of the risk assessment to manage allowed time in different plant configurations is complemented by the station's programs to monitor performance indicators for long-term availability of risk-significant components. The requirement to achieve acceptable long-term performance indicators provides an appropriate disincentive to the plant staff for regularly extending front-stop CT values to the detriment of safety risk and safety function availability.

RMTS-implementing plants must appropriately consider the issue of uncertainty when calculating configuration RICT values for plant-specific applications. However, the RICT quantitative acceptance guidelines established herein have the following fundamental basis. When RMTS-implementing plants apply PRAs of acceptable quality standards (see Section 5), application of PRA-calculated mean values (see definition in Appendix A) for configuration risk compared with the risk acceptance guidelines provided herein will meet acceptable uncertainty criteria for safe and prudent RICT implementation.

It is recommended that prior to implementation of the RMTS ("flexible CT"), a demonstration of the RI evaluation and control processes be performed. This demonstration may include limited post assessment of previous cycles' maintenance, or assessment of past NOEDs and a demonstration of how such situations would be handled when the RMTS process is instituted. In addition, a set of pre-defined failures of TS components can be postulated in the process of a normal maintenance schedule and the impact of delayed repair on plant risk and actions should be evaluated. Results of these studies may be used to inform the utility and NRC staff of the plant's program for implementing the flexible CT.

3.5.2.2 External events consideration

Plants without external events (Reference 20) PRAs must apply the following logic to support maintenance activities beyond the front-stop:

1. They must be able to provide a reasonable technical argument that the configuration risk of interest is dominated by internal events, and that external events (Reference 20) are an insignificant contributor to configuration risk, or

2. They must be able to perform a reasonable bounding analysis of the external events (Reference 20) contribution to configuration risk and apply this upper bound external events risk contribution along with the internal events risk contribution in calculating the configuration risk and the associated RICT, or

3. They must identify and implement risk mitigation or contingency actions that, for the duration of the configuration of interest, enable them to provide a reasonable technical argument that external events (Reference 20) are an insignificant contributor to configuration risk.

It is the intent of the RMTS process to consider the total plant risk. Plants with full scope PRAs may be able to largely perform quantitative risk assessments. However, it is expected that many of the plants intending to utilize the flexible CT will have robust level 1 PRAs and qualitative risk insights associated with fire, seismic and external flooding assessments. Checklists may be used to identify components where external event (Reference 20) overlaps are not significant and to limit maintenance in areas when the component risks are dominated by external event contributions.

3.5.3 Risk Management Actions

Determining actions, individually or in combinations, to control risk for a maintenance activity is specific to the particular activity (or maintenance configuration), its impact on risk, and the practical means available to control the risk. Some example actions are shown below:.

Normal work controls would be employed for configurations having predicted risk levels within RMTS lower-level thresholds (risk-informed safety criteria) presented in Table 3-2. This guidance means that the normal plant work control processes are followed for the maintenance configuration, and that no additional actions to address risk management are necessary.

Risk management actions, up to and including plant shutdown, should be implemented for plant configurations whose instantaneous and cumulative risk measures are predicted to approach or exceed lower-level RMTS thresholds. The benefits of these actions may or may not be easy to quantify. These actions are aimed at providing increased risk awareness of appropriate plant personnel, providing more rigorous planning and control of the maintenance activity, and taking steps to control the duration and magnitude of the increased risk. Examples of risk mitigation/management actions are as follows:

1. Actions to provide increased risk awareness and control:
    - Discuss planned maintenance activity and the associated maintenance configuration risk impact with operating shift crews and obtain operator awareness and approval of planned evolutions
    - Conduct pre-job briefing of maintenance personnel, emphasizing risk aspects of planned maintenance evolutions
    - Request/require that system engineer(s) be present for the maintenance activity, or for applicable portions of the activity
    - Obtain plant management approval of the proposed activity
    - Identify return-to-service priorities
2. Actions to reduce duration of maintenance activity:
    - Pre-stage required parts and materials accounting for likely contingencies
    - Walk-down the anticipated associated system tagout(s) and key equipment associated with the specified maintenance activity(ies) prior to conducting actual system tagout(s) and performing the maintenance
    - Develop critical activity procedures for risk-significant configurations, including identification of the associated risk and contingency plans for approaching/exceeding the RICT target.
    - Conduct training on mockups to familiarize maintenance personnel with the activity prior to performing the maintenance
    - Perform maintenance around the clock rather than "day-shift only"
    - Establish contingency plan to restore key out-of-service equipment rapidly if and when needed
3. Actions to minimize the magnitude of risk increase:

- Minimize other work in areas that could affect related initiating events (e.g., reactor protection system (RPS) equipment areas, switchyard, diesel generator (D/G) rooms, switchgear rooms) to decrease the frequency of initiating events that are mitigated by the safety function served by the out-of-service SSC
- Identify remain-in-service priorities and minimize work in areas that could affect other redundant systems (e.g., HPCI/RCIC rooms, auxiliary feedwater pump rooms), such that there is enhanced likelihood of the availability of the safety functions at issue served by the SSCs in those areas
- Establish alternate success paths (provided by either safety or non-safety related equipment) for performing the safety function of the out-of-service SSC
- Establish other compensatory measures as appropriate
- A final action threshold (i.e., a cumulative risk threshold) should be established such that plant staffs are discouraged from routinely and repeatedly entering risk significant configurations voluntarily.

Technical specifications LCO required actions, up to and including controlled plant shutdown, should be considered for plant configurations where instantaneous and cumulative risk measures are predicted to exceed upper-level RMTS thresholds presented in Table 3-2.. The plant RMTS program should include a clear decision process for determining when plant shutdown should be implemented as a result of maintenance configuration risk. An RMTS program shutdown decision process should include the following considerations:

- Evaluation of the projected integrated maintenance configuration risk (is it unacceptably high based on Table 3-2 thresholds?)
- Evaluation of the projected maintenance configuration duration and complexity (short and simple versus long and complex)
- Evaluation of the potential challenges to maintenance-affected SSCs imposed by a plant shutdown
- Evaluation of the alternative risk imposed by shutting the plant down (does the difference in integrated plant risk projected as a result of shutting down represent a significant "risk benefit" over the increased operational risk projected as a result of remaining at power?)

In this process, risk is "acceptable" when it is projected to remain within the upper-level RMTS thresholds (safety limit criteria) presented in Table 3-2.

3.6 Regulatory Treatment of Compensatory Measures

Using compensatory measures is discussed in several sections of this guide and in Reference 3. These measures may be employed, either prior to or during maintenance activities, to mitigate risk impacts. The following guidance discusses the applicability of 10 CFR 50.65 (a)(4) and 10 CFR 50.59 to the establishment of compensatory measures. There are two circumstances of interest:

1. The compensatory measures are established to address a degraded or nonconforming condition, and will be in effect for a time period prior to conduct of maintenance to restore

the SSC's condition. Per NRC Generic Letter 91-18, Revision 1 (and NEI 96-07, Revision 1), the compensatory measures should be reviewed under 10 CFR 50.59. If the compensatory measures are put into effect prior to performance of the maintenance activity, no immediate assessment is required under 10 CFR 50.65 (a)(4), however an assessment would be required prior to performing maintenance to address a degraded or nonconforming condition.

2. The compensatory measures are established as a risk management action to reduce the risk impact during a planned maintenance activity. The 50.65(a)(4) assessment should be performed to support the conduct of the corrective maintenance, and those compensatory measures that will be in effect during performance of the maintenance activity. The compensatory measures would be expected to reduce the overall risk of the maintenance activity; however, the impact of the measures on plant safety functions should be considered as part of the risk evaluation. Since the compensatory measures are associated with maintenance activities, no review is required under 10 CFR 50.59, unless the measures are expected to be in effect during power operation for greater than 90 days.

3.7 Documentation

The following are guidelines for documentation of the risk assessment:

1. Similar to 10 CFR 50.65 paragraph (a)(4) of the maintenance rule, the purpose of the RMTS program RICT assessment is to assess impacts on plant risk or key safety functions due to maintenance activities. This purpose must be affected through establishment of plant procedures that address process, responsibilities, and decision approach. It may also be appropriate to include a reference to the plant (a)(4) procedures and other appropriate plant procedures that govern planning and scheduling of maintenance or outage activities in the RICT assessment documentation. The RICT assessment process itself will be documented.

2. Also similar to (a)(4), it is not necessary to document the basis of each RICT assessment for removal of equipment from service as long as the RICT assessment process is followed. However, risk assessments and risk management actions for each entry into RMTS that exceeds the associated conventional technical specification "front stop" CT must be documented.

4
PRA ATTRIBUTES

The PRA used for the (a)(4) and RMTS risk assessments is important for two aspects:

1. Determination of scope of SSCs to which the assessment applies.

2. Evaluation of risk impact of the maintenance configuration (or as the basis for the risk monitor, matrix, or other tool), if the assessment is performed quantitatively.

In general, the quantitative risk assessment should be based on the plant Maintenance Rule risk assessment program (Reference 3) supported by the plant PRA. An internal-events-only Level 1 PRA may be applied if the containment breech and external events (Reference 20) risk associated with the configuration of interest can be shown to be reasonably bounded or insignificant via a blend of qualitative arguments and quantitative calculations.

The PRA model attributes and quality assurance requirements for RMTS applications are designed to be consistent and compatible with NRC regulatory guidance for PRA technical adequacy (see Reference 21), and supported by the ASME and ANS PRA Standards (References 19 and 20, respectively). Guidance provided in References 5, 16, and 34 also applies.

All portions of the PRA that support RICT assessments for SSCs within the scope of the RMTS program should be consistent with ASME PRA standard (Reference 19) requirements, as discussed in NRC draft regulatory guide DG-1122.

5 REFERENCES

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A
GLOSSARY OF TERMS

Key terms used in this guide are defined in this appendix. These definitions are intended to be consistent with existing plant technical specifications and associated regulatory and industry guidance. In any case where a plant's technical specifications definitions differ from those provided herein, the plant technical specifications definitions take precedence.

- *ACTION*: that part of a plant technical specification that prescribes remedial measures required under designated conditions.
- *AGGREGATE RISK*: the cumulative risk integrated over time accounting for variations in instantaneous risk; generally measured in terms of cumulative CDP and/or LERP (see definitions below).
- *ALLOWED OUTAGE TIME (AOT)*: the duration that an SSC specified in the plant technical specifications can be out of service (non-operational) during plant at-power operation before formal action is required via technical specification limiting conditions for operation.
- *AVERAGE RISK*: the average annual risk calculated via the plant PRA, accounting for the "average" or "typical" maintenance profile of the plant throughout the year. This is different from (generally greater than) the baseline "no-maintenance" risk of the plant.
- *BACK-STOP COMPLETION TIME*: the ultimate maintenance completion time or allowed outage time limit for a specified maintenance configuration. While 10CFR50.59 indicates that this limit may be reasonably established at 90 days, this guide has conservatively recommended a back-stop completion time of 30 days. The back-stop completion time limit for licensee action takes precedence over any risk-informed completion time calculated to be greater than 30 days.
- *BASELINE RISK*: the "no-maintenance" or "zero-maintenance" risk calculated via the plant PRA. This is different from (generally less than) the average annual risk calculated via the PRA.
- *COMPLETION TIME (CT)*: [Same as allowed outage time (AOT)]
- *CORE DAMAGE FREQUENCY (CDF)*: expected number of core damage events per unit of time.
- *CORE DAMAGE PROBABILITY (CDP)*: the integral of CDF over time; the classical cumulative probability of core damage (i.e., instantaneous core or fuel damage frequency integrated over a specified duration), over a given period of time. CDP is unit-less. Weekly risk is calculated for the 168-hour time period over each calendar week. Configuration risk is calculated for the anticipated and/or actual duration of a plant maintenance configuration. Annual risk is a 52-week rolling average, calculated week by week.
- *CUMULATIVE RISK*: same as "aggregate risk" defined above.

- *EMERGENT EVENT*: any condition, which is NOT in the planned work schedule, which renders station equipment non-functional or extends non-functional equipment scheduled outage time beyond its planned duration.
- *FRONT-STOP COMPLETION TIME*: the maintenance completion time or allowed outage time for plant equipment specified in the conventional (pre-RMTS) plant technical specifications.
- *FUNCTIONAL*: SSC is capable of performing its intended function for both normal and emergency operations required to mitigate plant risk as modeled in the plant-specific PRA.
- *INCREMENTAL CORE DAMAGE FREQUENCY (ICDF)*: the frequency above a "no-maintenance" baseline CDF (generally expressed in terms of events per calendar year) that one can expect a reactor fuel core-damaging event to occur for a nuclear power plant of interest.
- *INCREMENTAL CORE DAMAGE PROBABILITY (ICDP)*: the integral of ICDF over time; the classical cumulative probability of incremental core damage over a given period of time. ICDP is unit-less. <u>Weekly risk</u> is calculated for the 168-hour time period over each calendar week. <u>Configuration risk</u> is calculated for the anticipated and/or actual duration of a plant maintenance configuration. <u>Annual risk</u> is a 52-week rolling average, calculated week by week.
- *INCREMENTAL LARGE EARLY RELEASE FREQUENCY (ILERF)*: the frequency above a "no-maintenance" baseline LERF (generally expressed in terms of events per calendar year) that one can expect a large early release of radioactivity (as defined in Reference 4) from a reactor core-damaging event to occur for a nuclear power plant of interest.
- *INCREMENTAL LARGE EARLY RELEASE PROBABILITY (ILERP)*: the classical cumulative probability of incremental large early release of radioactivity over a given period of time. ILERP is unit-less. <u>Weekly risk</u> is calculated for the 168-hour time period over each calendar week. <u>Configuration risk</u> is calculated for the anticipated and/or actual duration of a plant maintenance configuration. <u>Annual risk</u> is a 52-week rolling average, calculated week by week.
- *INITIATING EVENT*: any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release. The scope of initiating events addressed in this guide include the full scope of those defined in References 19 and 20.
- *INSTANTANEOUS CORE DAMAGE FREQUENCY* (CDF_{inst}): the instantaneous expected core damage frequency resulting from continued operation in a specific plant mode and a given plant configuration (generally presented with units of events/year). In the context of a RMTS, this parameter would likely be calculated continuously and reported hourly or upon a change in value. This term is very similar to the "core damage frequency" term defined above, but the focus here is on a single point in time, and not on longer term averages typically applied when reporting CDF.

- *INSTANTANEOUS LARGE EARLY RELEASE FREQUENCY*: the instantaneous expected large early release frequency resulting from continued operation in a specific plant mode and a given plant configuration (generally presented with units of events/year). In the context of a RMTS, this parameter would likely be calculated continuously and reported hourly or upon a change in value.
- *KEY SAFETY FUNCTION*: any safety function of equipment included within the scope of technical specifications limiting conditions for operation.
- *LARGE EARLY RELEASE FREQUENCY (LERF)*: expected number of large early releases per unit of time.
- *LARGE EARLY RELEASE PROBABILITY (LERP)*: the classical cumulative probability of large early release of radioactivity (i.e., instantaneous large early release frequency integrated over a specified duration), over a given period of time. LERP is unit-less. <u>Weekly risk</u> is calculated for the 168-hour time period over each calendar week. <u>Configuration risk</u> is calculated for the anticipated and/or actual duration of a plant maintenance configuration. <u>Annual risk</u> is a 52-week rolling average, calculated week by week.
- *LIMITING CONDITION FOR OPERATION (LCO)*: Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met.

  A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

  (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

  (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.
- *MAINTENANCE CONFIGURATION*: the consolidated state of all plant SSCs with their associated individual states of functionality (i.e., either functional or non-

functional) and alignment (including surveillance inspections and testing alignments) identified. Consistent with the maintenance rule and associated NEI guidance (Reference 3), the concept of "maintenance configuration" also encompasses the existence of other activities or conditions (such as severe weather) that can materially affect plant risk. In the context of a RMTS program, some plants may wish to interpret a RMTS maintenance configuration definition to be generally limited to plant SSCs that have or could have associated technical specification LCOs (i.e., AOT or CT limits), and "functionality" is defined as "available to perform its associated safety function." A maintenance configuration definition can be expressed as a "truth table" for all appropriate SSCs that states the current state of functionality (yes or no) of each SSC. The universe of possible maintenance states includes the "no-maintenance" state where all SSCs are functional. See the following simple example of a maintenance configuration definition table:

PLANT SSC (TAG NUMBER)	FUNCTIONAL (AVAILABLE TO PERFORM ITS ASSOCIATED FUNCTION)
SSC00001	YES
SSC00002	NO
SSC00003	YES
. . .	. . .
SSCXXXXX	YES

In the context of this guide, there are two major types of maintenance configurations, planned and unscheduled maintenance. A planned maintenance configuration is one that is intentionally and deliberately pre-scheduled (i.e., in a weekly maintenance plan). An unscheduled maintenance configuration results from an unintentional, emergent situation (i.e., discovery of failure or significant degradation of an SSC within the scope of the RMTS program or a forced, unscheduled extension of previously-planned maintenance).

- *OPERABLE and OPERABILITY*: a system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).
- *OPERATIONAL MODE or MODE*: an operational mode (i.e., mode) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in plant technical specifications.
- *PRA-CALCULATED MEAN VALUE*: the mean value of a probability distribution for a key risk measure, such as CDP or LERP, calculated via the PRA uncertainty

analysis. This uncertainty analysis involves propagation of input data uncertainty through the PRA risk quantification process.

- *PROBABILISTIC RISK ASSESSMENT (PRA)*: a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment, PSA).
- *RISK-INFORMED COMPLETION TIME (RICT)*: a plant-specific SSC maintenance configuration CT or AOT calculated based on maintaining plant operation within allowed risk thresholds or limits (presented in Section 3 of this report) and applying a formally approved configuration risk management program and associated probabilistic risk assessment.
- *RISK-MANAGEMENT TECHNICAL SPECIFICATIONS (RMTS)*: a plant-specific set of configuration-based technical specifications, based on a formally approved configuration risk management program and associated probabilistic risk assessment, designed to supplement previous conventional plant technical specifications.
- *SAFE SHUTDOWN CONDITION*: the plant shutdown condition in any defined (known) plant shutdown mode where the reactor $K_{effective} < 0.99$.

B
BACKGROUND

B. Background

B.1 The Maintenance Rule – Technical Specification Nexus

Plant Technical Specifications were intended to provide time limits on inoperability of design basis components during various plant modes. These times were designated as Allowed Outage Times (AOTs) or Completion Times (CTs) within TS action statements. In practice, these limits were used to identify what level of maintenance would be done on those components. As refueling outages became shorter, these times were used to help establish the "at power" maintenance durations for design basis and safety related components. While a few selected high-risk maintenance combinations were prohibited by the TS (namely maintenance on redundant trains of the same system), no limitations were provided on non-TS components and most plant configurations were not directly restricted. In some instances, on-line maintenance was primarily based on compliance with the TS CTs, and at times implementing TS required actions resulted in operation in less than desirable plant configurations.

In an effort to improve plant maintenance practices in the nuclear industry, the NRC issued the Maintenance Rule (10CFR50.65) as its first risk informed performance based regulation. The regulation required the licensee to assess and manage risk, including the important contribution of Balance of Plant (BOP) non-safety systems. At the initial issuance of the rule, performance of a risk informed assessment was not required. In November 2000, the Maintenance Rule was amended with the addition of paragraph (a)(4). Paragraph (a)(4) of 10CFR50.65 explicitly required that plants assess and manage risk in the conduct of maintenance operations. This rule requires that a "risk assessment" be performed prior to voluntary entry into a maintenance configuration, or as soon as practical, upon entry into a non-voluntary maintenance condition. The guidance for satisfying the requirements of this rule provision is defined in Section 11 of NUMARC 93-01 (Reference 3) and has been endorsed by the NRC in RG 1.182 (Reference 28). These guidance documents were built, in part, on the Configuration Risk Management program developed as part of the CEOG pilot for RG 1.177. A companion risk-informed rule (10CFR50.59) change associated with evaluating "permanent" plant changes, became active in January 2001.

As a result of the difference in intent of the TS and the Maintenance Rule, the control of plant maintenance could be inconsistently treated. For example, the Maintenance Rule provides for a risk assessment prior to voluntary entry into a maintenance configuration, with the emergent (unplanned) work being evaluated as soon as practical. On the other hand, while the TS requires no risk assessment, operation within certain plant configurations is explicitly restricted, require

defined actions including plant shutdown, and is subject to rigid time restrictions. Furthermore, unlike the TS, the Maintenance Rule is silent on identification of plant conditions requiring plant shutdown.

The RMTS intends to meld the two processes together by supplementing the fixed interval CTs and prescribed actions in the current TS with a risk-informed alternative. This alternative establishes flexible CTs controlled by the Maintenance Rule, and shutdown/mode change actions established from a risk assessment process. Thus, TS actions will explicitly consider the contemporaneous plant risks in managing the plant configuration and while conducting restorative actions. The process for assessing plant risks will represent a blending of quantitative information and qualitative considerations.

B.2 Historical Evolution

10CFR50.36 (Reference 29) requires that the plant's design basis be maintained and that when the plant is outside that design basis, actions be taken to restore that design basis. Plant shutdown is included among the actions to be considered. The regulation has no explicit requirement or process for establishing allowable times for these actions or the associated restorative actions. As the TS evolved, deterministic insights, simplified risk insights, and judgment were used to establish CTs and actions. However, for the most part, the forced plant shutdown was considered a safe action if design basis compliance could not be restored. Therefore, a forced plant shutdown would be required, even when continued plant operation is the lower risk alternative. Later, the TS became increasingly standardized, culminating in the development of the Improved Standard TS (ISTS). The goal of the ISTS was to simplify the TS structure and clarify the TS language. In addition, the ISTS sought to remove conflicts that existed among TS actions and to rationalize some specific TS by integrating risk insights into the associated actions. While the ISTS resolved many of the initial problems with earlier TS, the actions and allowed outage times (or completion times) remained largely deterministically driven.

In 1993, the Electric Power Research Institute (EPRI) began development of the *PSA Applications Guide* (EPRI report TR-105396) to help utilities that own and operate nuclear power plants use their PRAs to improve plant safety and resource allocation. The *PSA Applications Guide* was completed in August 1995. In December 1995, with support from industry owners groups, EPRI published its *Guidelines for Preparing Risk-Based Technical Specifications Change Request Submittals* (EPRI report TR-105867). Also, in 1995, the NRC published its final policy statement on *Use of Probabilistic Risk Assessment Methods in Nuclear Activities* in the Federal Register. In 1997, the NRC developed draft regulatory guides (reg. guides) and associated draft standard review plan (SRP) sections related to risk-informed applications of nuclear power plant regulation. These draft reg. guides and SRP sections were reviewed, revised, and published as final reg. guides and SRP sections during the 1997-1999 time frame. Specifically, NRC reg. guide 1.177 provides NRC guidance on risk-informed technical specifications programs. Throughout the 1990s, the nuclear power industry has also developed and implemented 10 CFR 50.65, the "Maintenance Rule," and more recently implemented 10 CFR 65(a)(4), the maintenance configuration risk management portion of the Maintenance Rule (see Section 1). Also, over the past four years, the Nuclear Energy Institute

(NEI) has formed the Risk-Informed Technical Specification Task Force (RITSTF) and the Technical Specifications Working Group to address specific issues associated with the process of "risk-informing" plant technical specifications. This risk management guide was developed to supplement the *PSA Applications Guide* and current RITSTF efforts, and support utilities in effective and efficient development of risk-management technical specifications (RMTS) implementation programs.

Following industry feedback from a 1998 stakeholders meeting, the NRC recommended that the industry consider an initiative to risk inform the plant TS. In response to that initiative, several public meetings were held to identify the aspects of the TS that are amenable to a risk informed treatment. Based on these meetings, the NRC and industry have embarked upon an effort to globally risk inform several aspects of the current TS. The product to emerge from this effort is the RMTS. This effort is an outgrowth of the emergence of a "risk conscious" regulatory environment at the NRC and several years of regulatory experience in evaluating and implementing risk informed changes to the current generation of TS. As with the existing generation of TS, the criteria for entry into the associated TS will be defined inoperabilities of a TS System, Structure or Component (SSC). Retention of this structure will ensure that the RMTS is fully compatible with the requirements of 10CFR50.36 (Reference 29). However, it is envisioned that, once fully implemented, the maintenance related actions for non-TS SSCs will also follow the same risk assessment process.

C
RISK PROFILE EXAMPLES

This appendix provides some realistic examples of risk-versus-time profiles for a typical nuclear power generating unit. These examples have been developed via the STPNOC CRMP risk calculation and monitoring tool, the STPEGS Risk Assessment Calculator (RAsCal). Table C-1 shows some realistic plant risk-versus-time data for three typical nuclear power plant maintenance configuration transition profile examples.

TABLE C-1
EXAMPLE STPEGS RISK PROFILE DATA

Example Number	Maintenance Configuration	Start Time (Year 2003)	End Time (Year 2003)	Technical Specification Front-Stop CT (Hours)	Applicable Technical Specification Action	Instantaneous Incremental CDF (Events/Year)	Time to 1E-06 Incremental CDP (Hours)	Time to 1E-05 Incremental CDP (Hours)	Back-Stop CT (Hours)
1	CCA DGA EWA HHA	03/31 00:00	04/01 00:00	168	Restore CCA, EWA, and HHA to OPERABLE status within 7 days; restore DGA to OPERABLE status within 14 days; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	5.15E-05	170.12	1701.18	720
	CCA DGA EWA HHA HHC	04/01 00:00	04/01 01:00	1	Restore HHA or HHC to OPERABLE status within 1 hour; otherwise be in HOT STANDBY within 6 hours, HOT SHUTDOWN within the following 6 hours, and COLD SHUTDOWN within the subsequent 24 hours.	1.52E-04	57.84	578.43	720
	CCA DGA EWA HHA	04/01 01:00	04/02 12:00	168 Minus Elapsed Time (25 here) = 143	Restore CCA, EWA, and HHA to OPERABLE status within 7 days; restore DGA to OPERABLE status within 14 days; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	5.15E-05	170.12	1701.18	720
2	EWC	03/31 00:00	04/05 00:00	168	Restore EWC to OPERABLE status within 7 days; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	3.92E-05	223.51	2235.10	720
	AFD EWC	04/05 00:00	04/06 00:00	72	Restore AFD to OPERABLE status within 72 hours; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	1.35E-04	65.07	650.74	720
	AFD	04/06 00:00	04/06 22:00	72 Minus Elapsed Time (24 here) = 48	Restore AFD to OPERABLE status within 72 hours; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	1.33E-05	660.00	6599.96	720
3	EWC	03/31 00:00	04/04 00:00	168	Restore EWC to OPERABLE status within 7 days; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	3.92E-05	223.51	2235.10	720

Example Number	Maintenance Configuration	Start Time (Year 2003)	End Time (Year 2003)	Technical Specification Front-Stop CT (Hours)	Applicable Technical Specification Action	Instantaneous Incremental CDF (Events/Year)	Time to 1E-06 Incremental CDP (Hours)	Time to 1E-05 Incremental CDP (Hours)	Back-Stop CT (Hours)
	AFD EWC	04/04 00:00	04/05 00:00	72	Restore AFD to OPERABLE status within 72 hours; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	1.35E-04	65.07	650.74	720
	EWC	04/05 00:00	04/05 12:00	168 Minus Elapsed Time (120 here) = 48	Restore EWC to OPERABLE status within 7 days; otherwise be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.	3.92E-05	223.51	2235.10	720

A brief description of the maintenance configuration designators applied in Table C-1 is provided in Table C-2.

TABLE C-2
MAINTENANCE CONFIGURATION DESIGNATOR DESCRIPTIONS FOR TABLE C-1

Maintenance Configuration Designator	Description of Inoperable Equipment
AFD	Turbine-driven auxiliary feedwater pump (and unique function-supporting components)
CCA	Component Cooling Water pump/heat exchanger train A (and unique function-supporting components)
DGA	Standby diesel generator train A (and unique function-supporting components)
EWA	Essential Cooling Water ventilation fan train A (and unique function-supporting components)
EWC	Essential Cooling Water ventilation fan train C (and unique function-supporting components)
HHA	High head safety injection pump train A (and unique function-supporting components)
HHC	High head safety injection pump train C (and unique function-supporting components)

Listing multiple designators for one configuration simply means that the corresponding system functions/trains are simultaneously unavailable during that configuration.

Example 1 in Table C-1 indicates that the plant had planned maintenance for CCA, DGA, EWA, and HHA initially, and had entered that configuration, but that subsequently, an emergent condition developed wherein, during the planned maintenance configuration, the HHC function also became unavailable. In this example, the HHC function was recovered first; then planned maintenance for CCA, DGA, EWA, and HHA was completed, subsequently. Similarly, in Example 2 in Table C-1, maintenance was planned for EWC, but during that planned maintenance activity, the AFD function became unavailable, as an emergent condition. In this case, though, the plant was able to complete maintenance on the EWC function prior to recovering the AFD function. In effect, this action placed the plant in a safer configuration such that more time was available to address the emergent problem with the AFD function before any administrative or regulatory safety limits were breached. Finally, in Example 3 in Table C-1,

maintenance was planned for EWC, and during that planned maintenance activity, the AFD function became unavailable, as an emergent condition, as in example 2. However, in this case, the plant staff was able to quickly restore the AFD function, thus placing the plant in a safer condition to continue with the planned EWC maintenance.

Note that in Table C-1, the eighth column simply indicates how long, in hours, it will take to reach an incremental CDP value of 1.00E-05. As this time is based on the constant instantaneous incremental CDF value presented in column five of Table C-1, one can calculate the time to reach other values of incremental CDP (e.g., 1.00E-06) based on simple factor relationships. For example, if we wish to know how long it would take to reach an incremental CDP value of 1.00E-06 for the first configuration of example 1, we simply calculate one tenth of the time shown to reach 1.00E-05 (in this case, approximately 170 hours).

The Westinghouse Owners Group has calculated maintenance risk profiles for example scenarios 1 and 2 in Table C-1 for some typical generic pressurized water reactor designs. The results of these calculations for the most limiting maintenance configuration of these two scenarios are presented in Tables C-3 and C-4 for example scenarios 1 and 2, respectively. Tables C-1, C-3, and C-4 show that, for typical, but challenging, maintenance configurations, reasonable time periods are available to plant staffs for prudent risk management action based on the RMTS quantitative risk acceptance guidelines presented in Table 3-2 of this report.

TABLE C-3
EXAMPLE SCENARIO 1 RISK PROFILE DATA FOR GENERIC PRESSURIZED WATER REACTOR TYPES

Generic Plant Type	Maintenance Configuration (see Table C-2)	Instantaneous Incremental CDF (Events/Year)	Time to 1E-06 Incremental CDP (Hours)	Time to 1E-05 Incremental CDP (Hours)	Back-Stop CT (Hours)	Plant Design Remarks
CE Early Design	CCA DGA EWA HHA HHC	5.99E-05	146.34	1463.44	720	Plants have diesel-driven startup feedwater pumps.
CE Later Design	CCA DGA EWA HHA HHC	1.99E-04	44.05	440.50	720	Plants have no PORV.
Westinghouse 2-Loop	CCA DGA EWA HHA HHC	2.21E-04	39.67	396.65	720	Plants have available non-safety equipment to support the auxiliary feedwater function.
Westinghouse 3-Loop	CCA DGA EWA HHA HHC	6.82E-04	12.85	128.53	720	None.

TABLE C-4
EXAMPLE SCENARIO 2 RISK PROFILE DATA FOR GENERIC PRESSURIZED WATER REACTOR TYPES

Generic Plant Type	Maintenance Configuration (see Table C-2)	Instantaneous Incremental CDF (Events/Year)	Time to 1E-06 Incremental CDP (Hours)	Time to 1E-05 Incremental CDP (Hours)	Back-Stop CT (Hours)	Plant Design Remarks
CE Early Design	AFD EWC	4.98E-05	176.02	1760.24	720	Plants have diesel-driven startup feedwater pumps.
CE Later Design	AFD EWC	1.05E-04	83.49	834.86	720	Plants have no PORV.
Westinghouse 2-Loop	AFD EWC	1.33E-04	65.91	659.10	720	Plants have available non-safety equipment to support the auxiliary feedwater function.
Westinghouse 3-Loop	AFD EWC	1.01E-04	86.79	867.92	720	None.

Exelon Nuclear

Presentation to NRC – Risk Informed Initiative 5b

November 13, 2003

Enclosure 7

Agenda

Exelon Nuclear

- Risk Informed Initiative 5b – Gene Kelly
- Initiative 5b Methodology/Guidance – Gene Kelly
- Status of Pilot Project – Phil Tarpinian
- STI #1 – CRD Testing – Phil Tarpinian
- Scope of February 2004 Submittal – Glenn Stewart
- Next Steps – Glenn Stewart

Gene

Risk Informed Initiative 5b

Exelon Nuclear

- Relocate surveillance intervals to licensee controlled document (e.g., TRM)
- Retain surveillance requirements in the Tech Specs
- Change the interval based on a risk informed process tempered by actual performance and existing commitments
- Interval changes would occur without prior NRC approval as long as the risk informed process is followed
- BWROG is lead Owners' Group
- Pilot Plants: Limerick and Peach Bottom

Gene

Initiative 5b Methodology/Guidance

Exelon Nuclear

- Combined NEI/BWROG document
- Utilized by the pilot plants
- Involves qualitative and quantitative reviews
- Considers risk significance of systems/components (high or low)
- Addresses PRA modeled and unmodeled systems/components

Phil

Status of Pilot Project

Exelon Nuclear

- Integrated Decisionmaking Panel (IDP) established
- IDP Charter developed
- ST interval extension evaluation template form developed
  - Based on NEI/BWROG methodology
- Commitment reviews performed
- Equipment and surveillance history evaluated
- Pilot ST intervals selected
  - CRD exercise testing
  - SGTS/RERS flow testing
  - 4 kV Undervoltage Relay testing
  - LOCA/LOOP testing
  - Turbine Valve/MSIV testing
- All legs of methodology tested

Exelon Nuclear

STI #1 – CRD Testing

- Rod notch testing
- Extend from weekly to monthly
- Qualitative review only
- Evaluation template reviewed by IDP
- Methodology path reviewed by IDP
- IDP approved extension conditionally
  - Pending resolution of commitment
- Monitoring and phase-in of proposed STI

Glen

Scope of February 2004 Submittal

Exelon Nuclear

- February 2004 submittal expected to include all of the following requests:
  - Relocate ST intervals to a licensee controlled document
  - Add a Surveillance Frequency Control Program to Tech Specs
  - Approval of candidate ST interval extensions which test the methodology
- Limerick submittal first, followed by Peach Bottom

Glen

Next Steps

Exelon Nuclear

- Complete remaining candidate ST interval extension reviews
- Invite NRC, NEI and BWROG to observe IDPs
- Finalize submittal, methodology, TSTF
- In February 2004:
  - Submit pilot plant License Amendment Request to NRC (Exelon)
  - Submit Initiative 5b methodology to NRC (NEI)
  - Submit TSTF-425 to NRC (BWROG)

Risk-Informed Technical Specifications Initiative 5B - Relocation of Surveillance Test Intervals

1.0 Introduction

The Technical Specification Task Force has established TSTF-425, which proposes that the Technical Specification Surveillance Frequency requirements would be relocated to a licensee controlled program. The surveillance requirement would still remain in the Technical Specifications, pursuant to 10 CFR 50.36. The Administrative Controls section of the Technical Specifications would specify the requirements for a Technical Specification Surveillance Frequency Control program that the licensee would use to control changes to the surveillance performance frequency.

This document proposes a risk-informed process and methodology for control of changes to the Technical Specification surveillance performance frequency for structures, systems, and components, (SSC), under the required licensee controlled program. The methodology of this document, once approved by NRC, provides the basis for changing the Technical Specification surveillance intervals within the required licensee controlled program.

The regulatory process for obtaining of the License Amendment approval from the NRC is beyond the scope of this document. Such details are contained in the Technical Specification Task Force TSTF-425.

NRC is requested to approve that this process and methodology provides a sufficient basis for the required licensee-controlled program of the relocated Technical Specification surveillance frequency requirements.

2.0 Overall Approach

The proposed approach uses existing Maintenance Rule implementation guidance (NUMARC 93-01 Rev. 3) (Reference 2), risk categorization guidance originally developed for the proposed 10 CFR 50.69, "Option 2" rulemaking, (NEI-00-04) (Reference 3), combined with elements of NRC In-service Testing Regulatory Guide 1.175 (Reference 6), to develop risk-informed test intervals for SSCs having Technical Specification surveillance test requirements. While originally developed to address test intervals for pump and valve testing required by the ASME Code, the concepts of Regulatory Guide 1.175 are applicable to this Technical Specifications Initiative with minor modifications. In particular, this Regulatory Guide provides information relative to modeling the effect of the revised surveillance intervals in the PRA.

The method described here is also consistent with Regulatory Guides 1.174 (Reference 4), "An Approach for Using Probabilistic Risk Assessments in Risk-Informed Decisions on Plant-specific Changes to the Licensing Basis" and 1.177 (Reference 5), "An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications"

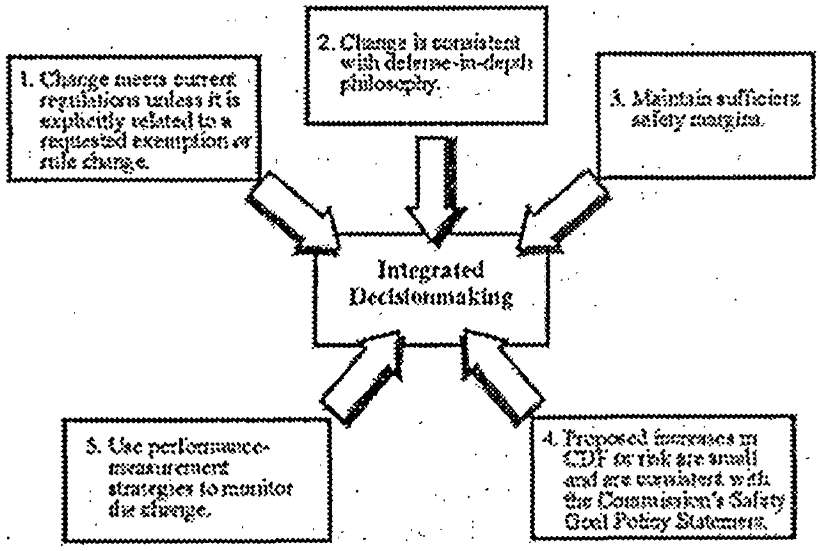
and provides more specific guidelines to facilitate application by the Licensee. Regulatory Guide 1.177 provides guidance for changing STI and Allowed Out-of-service Times (AOT). However, for allowable risk changes associated with STI extensions, it refers to RG 1.174, which provides quantitative risk acceptance guidelines for changes to core damage frequency (CDF) and large early release frequency (LERF). RG 1.174 provides additional guidelines that have been adapted for this methodology.

The detailed process is described in Section 4. The scope of evaluation includes those structures, systems and components (SSCs) having Technical Specification surveillance test interval requirements. PRA technical adequacy would be addressed through NRC draft Regulatory Guide DG-1122 (Reference 7) (or the final version of this Regulatory Guide). Following the establishment of adequate PRA capability, the process involves a categorization of the SSCs into high safety significance (HSSC), or low safety significance (LSSC). Once categorized, revised surveillance test intervals are developed based on risk insights from PRAs, plant operational experience, and other factors. For HSSCs, the PRA is used to quantify the effect of the proposed change, and the aggregate risk impact of the revised intervals is compared to NRC risk acceptance guidelines. For LSSCs, very low risk impact will result from the revised intervals, and no additional monitoring would be proposed, beyond those already being conducted under the Maintenance Rule. Feedback and periodic re-evaluation of the intervals will be conducted for HSSCs and LSSCs.

3.0 Key Safety Principles

Regulatory Guide 1.174 identifies five key safety principles to be met for all risk-informed applications and to be explicitly addressed in risk-informed plant program change applications.

Figure 1 of Regulatory Guide 1.174, illustrates the consideration of each of these principles in risk-informed decision making.



1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

10 CFR 50.36(c) provides that Technical Specifications will include items in the following categories:

"(3) *Surveillance Requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met".

Technical Specifications Initiative 5B proposes to relocate the intervals for the performance of the above requirements to a licensee-controlled program using an NRC approved methodology for control of the surveillance intervals. The surveillance requirements themselves would remain in Technical Specifications. Thus, this proposed change meets criterion 1 above.

2. The proposed change is consistent with the defense-in-depth philosophy.

Consistency with the defense-in-depth philosophy is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A is maintained.

These defense-in-depth objectives apply to all risk-informed applications, and for some of the issues involved (e.g., no over-reliance on programmatic activities and defense against human errors), it is fairly straightforward to apply them to this proposed change. The use of the multiple risk metrics of CDF and LERF and controlling their change resulting from the implementation of this initiative would maintain a balance between prevention of core damage, prevention of containment failure, and consequence mitigation. Redundancy, diversity, and independence of safety systems are considered as part of the risk categorization to ensure that these qualities are not adversely affected. Independence of barriers and defense against common cause failures are also considered in the categorization. The improved understanding of the relative importance of plant components to risk resulting from the development of this program should promote an improved overall understanding of how the SSCs contribute to a plant's defense in depth, and this should be discussed in the application.

3. The proposed change maintains sufficient safety margins.

Conformance with this principle is assured since Codes and Standards or alternatives approved for use by the NRC will still be met with the proposed changes. Also, the safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met with the proposed changes.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

In Initiative 5B, information obtained from a PRA is used in two ways: First, to provide input to the categorization of SSCs into HSSC and LSSC groupings; and second, to assess the impact of the proposed change on CDF and LERF. The overall impact of the change is assessed and compared to the quantitative risk acceptance guidelines of Regulatory Guide 1.174, which is consistent with the intent of the Commission's Safety Goal Policy Statement. More detail is provided in subsequent paragraphs that describe the process.

The PRA used to support this change will, at a minimum, address core damage frequency and large early release frequency for power operation. External event risk and shutdown considerations will be addressed through quantitative or qualitative means.

NRC Draft Regulatory Guide DG-1122 addresses technical adequacy of PRA for risk-informed applications. This regulatory guide (in final form) will be followed for plants proposing to implement Initiative 5B.

5. The impact of the proposed change should be monitored using performance measurement strategies

For HSSCs, a performance monitoring strategy should be developed to provide confidence that the equipment performance is consistent with the considerations of the overall process, and is not degrading such that the analysis assumptions and expert panel judgments are no longer valid. The output of the performance monitoring should be periodically re-assessed, and appropriate adjustments made to the test intervals.

LSSCs have been shown through a robust categorization process (NEI-00-04) to be of low risk significance. Changes to surveillance test intervals for these SSCs will have a small effect on their performance, and thus will have a second order effect on risk. Because of this minimal risk impact, existing performance monitoring required by the Maintenance Rule is adequate for LSSCs whose surveillance intervals are addressed by Initiative 5B.

4.0 STI Revision Process

The process is shown in a flow diagram in the Figure 1. The process steps are described below:

4.1 Steps 1 through 4 – Categorize SSCs into High Safety Significance (HSSC) or Low Safety Significance Components (LSSC).

Step 1: Existing Maintenance Rule Categorization

The Maintenance Rule also addresses SSCs that are subject to Technical Specification surveillance test requirements. The Maintenance Rule requires that licensees monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. Such goals are to be established, where practical, commensurate with safety, and they are to take into account industry wide operating experience. When the performance or condition of a component does not meet established goals, appropriate corrective actions are to be taken.

Implementation guidance for the Maintenance Rule has been developed and approved by NRC. This guidance, NUMARC 93-01, Revision 3, provides that:

1. Insights from probabilistic risk assessment (PRA) be used to determine the risk significance of affected SSCs, through the use of risk importance measures.
2. SSC availability and reliability impacts should be balanced in a manner that addresses the risk insights from the PRA.
3. Performance monitoring of SSCs should be conducted commensurate with their risk impact

Step 2: Recategorize HSSC?

As noted in the discussion relating to the previous step, the Maintenance Rule provides a basis for classification of SSCs as either HSSC or LSSC. Licensees implementing Initiative 5B may choose to retain the existing Maintenance Rule classification for Technical Specification SSCs currently classified as HSSC. Otherwise (e.g., for Maintenance Rule LSSCs, or for potential recategorization of Maintenance Rule HSSCs as LSCCs), the NEI-00-04 categorization process should be followed. For many SSCs that are obviously of high-risk importance, the decision to retain the existing HSSC designation is an efficient approach.

The categorization may be conducted on a functional level or on an SSC level, as discussed in NEI-00-04. This is discussed in detail in Step 4

Step 3: DG-1122 PRA Technical Adequacy

NRC is developing regulatory guidance to address PRA technical capability. Currently this is a draft Regulatory Guide, DG-1122, (Reference 7). This regulatory guide addresses the use of the ASME PRA standard, and the NEI peer review process (NEI-00-02) for evaluating PRA technical capability.

DG-1122 also provides attributes of importance for risk determinations relative to external events, seismic, internal fires, and shutdown.

It is envisioned that plants implementing initiative 5B would evaluate their PRA in accordance with this regulatory guide. The regulatory guide specifically addresses the need to evaluate important assumptions that relate to key modeling uncertainties (such as reactor coolant pump seal models, common cause failure methods, success path determinations, human reliability assumptions, etc). Further, the regulatory guide addresses the need to evaluate parameter uncertainties and demonstrate that calculated risk metrics (e.g, CDF and LERF) represent mean values.

This step is shown in dotted line since this is actually related to the adequacy of the process itself, and getting the process ready for the evaluation, rather than the impact of the STI change.

Step 4: NEI-00-04 Categorization

NEI-00-04 addresses all necessary considerations for categorization of components for proposed 10 CFR 50.69, as well as this application. This document provides for an expert panel process using insights from available risk information and includes consideration of the following:

- Internal events risk bases on a PRA
- Fire risk using a Fire PRA or FIVE analysis
- Seismic risk using a seismic PRA or seismic margins analysis
- Shutdown risk using a shutdown PRA or shutdown risk studies
- Use of risk importance measures
- Components not modeled in the PRA
- Sensitivity Studies

NEI-00-04 will be followed, unless the Licensee determines that current Maintenance Rule HSSC categorizations will be maintained for this application. NEI-00-04 contains a final sensitivity study, specific to the §50.69 rulemaking, that involves raising the failure rates of all RISC-3 (safety related but low safety significance) SSCs by a specific factor. This portion of NEI-00-04 is not applicable to the Technical Specifications 5B initiative, and the overall risk impact of this initiative will be demonstrated through other means as discussed later in this paper. *[Note, plants also implementing Option 2 and desiring a consistent process (and result) for SSC categorization for all applications would need to use the NEI-00-04 final sensitivity study to meet the categorization requirements for initiative 5B.]*

4.2 Steps 5 through 7 – Process for SSCs Categorized as Low Safety Significant (LSSC)

Step 5: Identify Qualitative Considerations to be Addressed

The Expert Panel selects the STI intervals for the LSSC systems based on qualitative consideration. (See additional details on Expert Panel in Step 6 and 22). In Step 5, such qualitative considerations are developed as an input to the Expert Panel. Such considerations include, but are not limited to:

- Surveillance test and performance history of the components and system associated with the STI extension
- Uncertainty associated with the quantitative process
- The impact of systems not quantified using the internal event PRA
- The impact of systems for which LERF results are not available

- The impact of systems for which external events and shutdown PRA are not available
- Past industry and plant-specific experience with the functions affected by the proposed changes
- Impact on defense-in-depth protection.
- Vendor-specified maintenance frequency
- ASME and other code-specified test intervals

The qualitative considerations relative to the proposed STI changes are presented to the Expert Panel (Step 6).

Step 6: Expert Panel Determines New STI for the LSSC

This step involves the use of an Expert Panel, which is charged with the task of reviewing the STI extensions qualitatively. The details on the constitution of the Expert Panel are covered in Step 22.

The Expert Panel reviews and approves the revised STI for the LSSC systems based on factors such as operating history, reliability and availability.

After the Expert Panel approves the revisions, the changes are implemented and documented for future audits by NRC. If the Expert Panel does not approve certain STI extension, then the STI value is left unchanged.

Step 7: Document New STI and Implement the Changes

The STI changes approved by the Expert Panel for the LSSC are documented appropriately and then implemented by revising plant procedures, document and training the personnel as needed. The STI change process stops here, however, long monitoring is still required per Step 25.

4.3 Steps 8 through 11 Checking Commitments for HSSC

Step 8: Check for Prohibitive Commitments

In Step 8, all the commitments made to the NRC relative to the HSSC system are collected and reviewed. Some of the commitments to maintain a certain surveillance test interval may have been made in relation to certain other plant issues. As part of this step, such commitments are identified and then in Step 9 it is examined if the commitment itself can be changed. If there are no such commitments, then STI change process continues in Step 12

Step 9: Can Commitments be Changed?

In Step 9, a check is made to determine if the NRC commitments can be changed. If the commitment can be changed, go to Step 10 for changing the commitment.

If the commitment cannot be changed, document the information and leave the STI unchanged in Step 11.

Step 10: Change the Commitment

In Step 10, change the commitment using NRC approved process such that the STI can be revised using this process. Changing the NRC commitment is a separate activity. Return to the process after the commitment has been changed. If the NRC does not permit commitment change, go to Step 11 since STI change cannot be implemented. After the commitment change has been made, go to step 12.

Step 11: Document that STI Changes Cannot be Changed

This step is entered if in Step 9 it is determined that the STI cannot be changed. Document that STI cannot be changed and the process concludes here.

4.4 Steps 12 through 22 – Process for SSCs Categorized as High Safety Significant Components (HSSC)

Step 12: Select Desired Revised STI Values

Technical Specifications STIs are identified for improvement. This identification is done based on the difficulty of the test, cost of the test, potential for error during the test and its consequence, and the role of the test on the reliability of the associated function. The Licensee should also identify the desired revised STI values.

Following this step, the process diverges into two paths, both of which need to be followed. One of paths, starting at Step 13 performs a qualitative evaluation and the other one starting at Step 14 leads to a quantitative evaluation. Both paths converge later at Step 22.

Step 13: Identify Qualitative Considerations to be Addressed

Qualitative considerations are developed as an input to the Expert Panel. Such considerations include, but are not limited to:

- Surveillance test and performance history of the components and system associated with the STI extension
- Uncertainty associated with the quantitative process
- The impact of systems not quantified using the internal event PRA
- The impact of systems for which LERF results are not available
- The impact of systems for which external events and shutdown PRA are not available

- Past industry and plant-specific experience with the functions affected by the proposed changes
- Impact on defense-in-depth protection.
- Vendor-specified maintenance frequency
- ASME and other code-specified test intervals

The qualitative considerations are presented to the Expert Panel (Step 22) along with the quantitative considerations from Step 21.

Step 14: Associated STI SSC Modeled in PRA?

Check if the surveillance tests or the associated system or component is modeled in the PRA. At this point, the focus is on the internal event full power PRA, although the question is applicable for the external event and shutdown PRA as well. If yes, go to Step 18. If not, go to Step 15 to determine if STI can be modeled in the PRA.

The process requires that, as a minimum, the internal event full power PRA be available. However, if the Fire, Seismic or Shutdown PRA is not available, then go directly to Step 16 to carry our bounding analysis for that PRA, but continue with the process.

Step 15: Can STI Be Modeled in PRA?

Step 15 is entered if in Step 14, it is determined that the associated STI system or component is not modeled in the PRA. Here the analyst has to decide if the STI can be modeled in the PRA. The determination pertains to all PRAs, including external events and Shutdown, but the initial focus is on the internal event PRA. If STI can be modeled in the PRA, go to Step 17. If not, go to Step 16.

Step 16: Perform Bounding Risk Analysis

Step 16 is entered from Step 15 where it is determined that STI extension cannot be modeled in the plant PRA. In such a case, the PRA analyst will have to perform bounding analyses that would provide some indication of the impact of the STI extension on the PRA results. Bounding analyses are either quantitative analysis carried out with available PRA models or qualitative evaluation using deterministic considerations. Results of the analyses are sent to the Expert Panel in Step 22.

Step 17: Revise PRA Model as Needed

Step 17 is entered from Step 15 where it is determined that STI extension can be modeled in the PRA. Modify the PRA to reflect the STI extensions. Section 2.3.3 of RG 1.175 provides guidance on PRA modeling. It states that the assumption that the total unavailability scales linearly with the test interval is conservative

and is acceptable to NRC. However, for more realistic modeling and to justify longer STIs, modeling the "demand" contribution in addition to the test-interval-dependent contribution to system unavailability would be needed. The output of this Step is an input to the Step 18, CDF and LERF evaluation.

Step 18: Evaluate Cumulative Effect on CDF & LERF

In Step 18, the cumulative effect on CDF and LERF of all Risk-informed STI revisions on all PRAs (internal event, fire, flood, seismic event, and shutdown) is evaluated.

Step 19: Total CDF & LERF Change <1.174 Limits?

In Step 19, the cumulative impact of all Risk-informed STI extensions on all PRAs (internal event, fire, flood, seismic event and shutdown) must also meet the RG 1.174 limits for CDF and LERF changes. If the RG 1.174 guidelines (limits) are met, then go to Step 21. If not, go to Step 20 where the proposed STI values are revised and sent for reevaluation (to Step 18). The CDF and LERF values are re-evaluated. (Note: The sequence of Steps 18, 19, and 20 are repeated until at Step 19, CDF and LERF values are determined to be within the RG 1.174 limits).

Step 20: Revise STI Values

Step 20 is entered where it is determined that STI revisions do not meet the Regulatory Guide 1.174 acceptance criteria (Step 19), or are not supported by sensitivity studies (Step 21). The STI(s) are adjusted accordingly and re-evaluated in Step 18.

Step 21: Perform Sensitivity Studies

Carry out risk sensitivity studies by changing the unavailability terms for PRA basic events that corresponds to SSCs being evaluated. As stated in Section 8 of NEI-00-04, the basic events for both random and common cause failure events should be increased for failure modes impacted by the changes. A factor of 2 to 5 is appropriate as sensitivity because it is representative of the change in reliability between a mean value and an upper bound (95th percentile) for typical equipment reliability distributions. For example, for a lognormal distribution the ratio of 95th percentile to mean value would be approximately 2.4 for an error factor of 3 and 3.5 for an error factor of 10.

Other issues that should be addressed in the quantification of the change in risk include the following.

- The impact of the surveillance interval change on the frequency of event initiators (those already included in the PRA and those screened out because of low frequency) should be determined. For applications in this

initiative, potentially significant initiators include valve failure that could lead to interfacing system loss-of-coolant accidents (LOCAs) or to other sequences that fail the containment isolation function.

- The effect of common cause failures (CCFs) should be addressed either by the use of sensitivity studies or by the use of qualitative assessments that show that the CCF contribution would not become significant under the Initiative 5B program (e.g., by use of phased implementation, staggered testing, and monitoring for common cause effects).
- Justification of surveillance interval changes should not be based on credit for post-accident recovery of failed components (repair or ad hoc manual actions, such as manually forcing stuck valves to open). However, credit may be taken for proceduralized implementation of alternative success strategies. The evaluation should be performed so that the truncation of LSSCs is considered. It is preferred that solutions be obtained from a re-solution of the model, rather than a reunification of CDF and LERF cutsets.

If the sensitivity evaluation shows that the changes in CDF and LERF as a result of changes in SSCs being evaluated are not within the acceptance guidelines of Regulatory Guide 1.174, than revised STI(s) may be needed (go to Step 20). If the sensitivity evaluation supports the STI changes, then go to Step 22.

Step 22: Expert Panel Approval or Adjust STI

This step involves the use of an Expert Panel, which in addition to reviewing the results quantitatively, is charged with the task of reviewing the STI extensions qualitatively.

The qualifications for the Expert Panel members are very similar to the one for the Maintenance Rule. Normally the same Expert Panel, (referred to as IDP, Integrated Decisionmaking Panel), is used as for the Maintenance Rule implementation. A specialist with experience in surveillance tests and system or component reliability should also be added to the Expert Panel.

If the Expert Panel approves the change, the changes are implemented and documented for future audits by NRC. If the Expert Panel does not approve certain STI extension, then the STI value is left unchanged.

The Expert Panel has additional responsibilities. These relate to making recommendations on the way the revised surveillance intervals are implemented (for instance, a phased implementation), reviewing the cumulative impact of all changes carried out over a period of time, and monitoring the impact of changes on failure rates.

Step 23: Document New STI and Implement the Changes

This step is similar to Step 7. The STI changes approved by the Expert Panel for the HSSC are documented appropriately and then implemented by revising plant procedures, document and training the personnel as needed. The STI change process stops here, however, long-term monitoring is still required per Step 24.

4.5 Steps 24 through 27 – Long-Term Monitoring and Feedback

Step 24: Monitoring & Feedback

The purpose of performance monitoring in this process is twofold. First, performance monitoring should help confirm that no failure mechanisms that are related to the revised test strategies become important enough to alter the failure rates assumed in the justification of program changes. Second, performance monitoring should, to the extent practicable, ensure that adequate component capability (i.e., margin) exists relative to design-basis conditions, so that component-operating characteristics, over time, do not result in reaching a point of insufficient margin before the next scheduled test activity. Regulatory Guide 1.174 (Ref. 4) provides guidance on performance monitoring when testing under design basis conditions is impracticable. Component or train level monitoring would be expected for HSSCs.

Two important aspects of performance monitoring are whether the test frequency is sufficient to provide meaningful data and whether the testing methods, procedures, and analysis are adequately developed to ensure that performance degradation is detected. Component failure rates should not be allowed to rise to unacceptable levels (e.g., significantly higher than the failure rates used to support the change) before detection and corrective action take place.

For acceptance guidelines, monitoring programs should be proposed that are capable of adequately tracking the performance of equipment that, when degraded, could alter the conclusions that were key to supporting the acceptance of the initiative 5B program. Monitoring programs should be structured such that SSCs are monitored commensurate with their safety significance. This allows for a reduced level of monitoring of components categorized as having low safety significance.

The performance monitoring process should have the following attributes:

- Enough tests are included to provide meaningful data,
- The test is devised such that incipient degradation can reasonably be expected to be detected.

The output of this step is sent to Step 26.

Step 25: Existing Maintenance Rule (or 10CFR 50.69) Monitoring

For LSSCs, very low risk impact is expected from the revised intervals, and no additional monitoring is proposed, beyond that already conducted under the Maintenance Rule. Feedback and periodic re-evaluation of the intervals will be conducted for HSSCs and LSSCs (Step 26).

Step 26: Periodic Re-assessment

The Initiative 5B program contains provisions whereby component performance data periodically is fed back into the component test strategy determination (i.e., test interval and methods) process. This would include results of component or train level monitoring for HSSCs, and results of Maintenance Rule (or §50.69 monitoring) for LSSCs.

Measures should be in place to identify the need for more emergent program updates (e.g., following a major plant modification or following a significant equipment performance problem). The results of these periodic re-assessments are fed back to the Expert Panel in Step 27 for evaluation.

Step 27: Expert Panel Reviews Experience Feedback

Step 27 is entered from Step 26 where the operating experience feedback following STI change implementation is reviewed periodically.

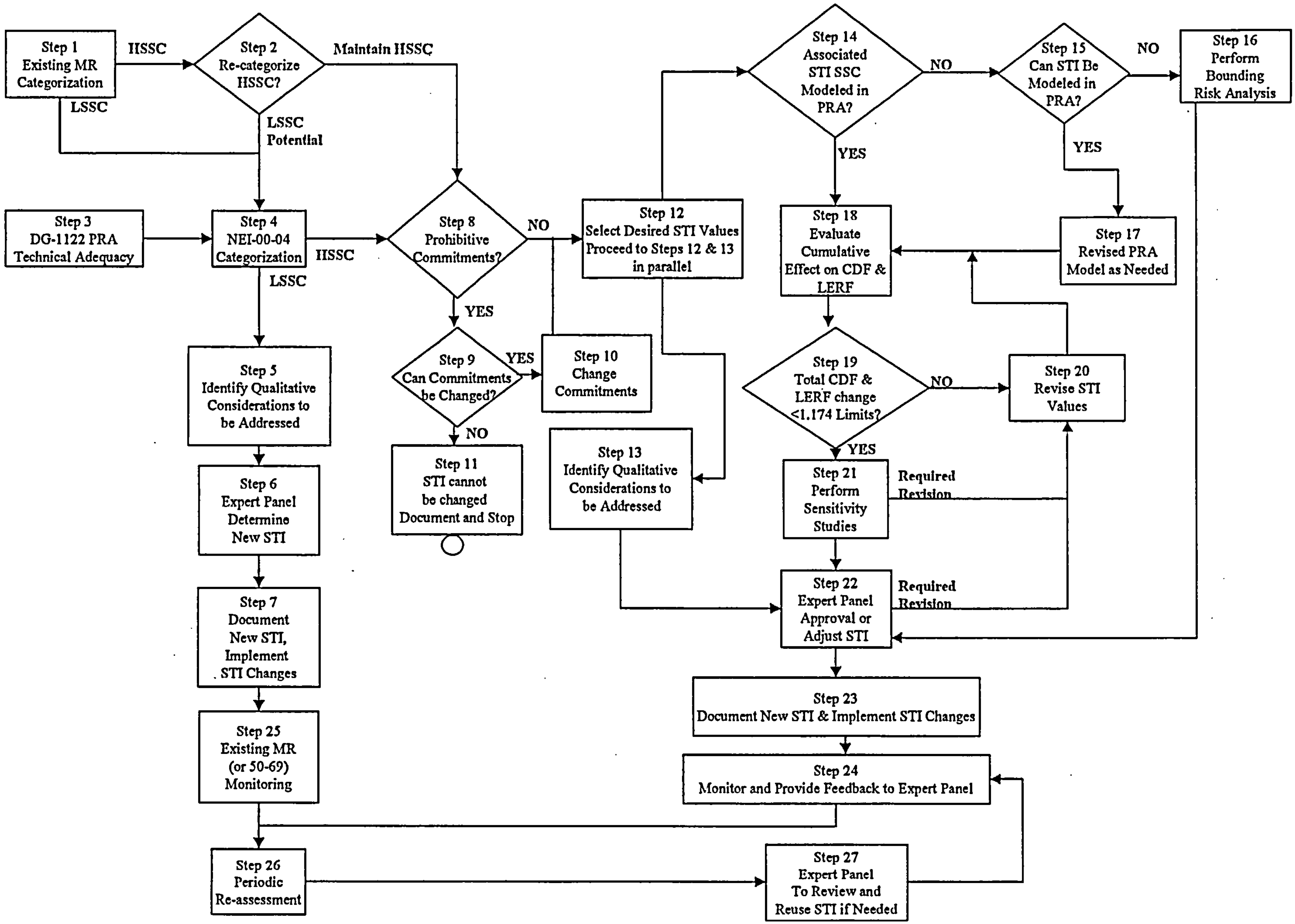
The Expert Panel would be responsible for periodic review of performance monitoring results (from Step 26) and attendant re-assessment of the Initiative 5B program. Any changes identified by the Expert Panel are routed to Step 24, Revise STI Values and continue to monitor the results.

5.0 References

1. 10 CFR50.69, *"Scope of Structures, Systems and Components, Governed by Special Treatment Requirements"*
2. NUMARC 93-01, Rev. 3 *"Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"*
3. NEI 00-04 (Draft), *10 CFR 50-69 SSC Categorization Guideline*, June 2002
4. Regulatory Guide 1.174, *"An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*" July 1998
5. Regulatory Guide 1.177, *"An Approach for Plant-specific Risk-informed Decisionmaking: Technical Specification"*, August 1998
6. Regulatory Guide 1.175, *"An Approach for Plant-Specific Risk-Informed Decisionmaking: Inservice Testing", August 1998.*
7. Draft Regulatory Guide DG-1122, *"An approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities", November 2002.*

Figure 1

Tech Spec Initiative 5 B Process



IOY Copy : STI #2

RI-TS Surveillance Test Interval (STI) Evaluation

BWROG RI-TS Initiative 5b Pilot (Exelon BRIM 132)

Station: Limerick Generating Station **Unit(s):** Units 1 and 2

Surveillance Test Number (s): ST-6-076-250-1(2), SGTS and RERS Flow Test **Revision Number:** Unit 1-Rev. 34 / Unit 2- Rev. 26

Technical Specification Surveillance Requirement (SR) Number(s): 4.6.5.3.a and 4.6.5.4.a
Technical Specification SR (Text):

4.6.5.3.a Each standby gas treatment subsystem shall be demonstrated OPERABLE:
At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

4.6.5.4.a Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:
At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly.

Technical Specification SR Bases (and Intent): The specific basis for the 31 day surveillance interval is not described in the LGS TS Bases. The basis for this monthly test is related to the elimination of moisture in the filters and adsorbers as described in Regulatory Guide 1.52.
Recommended STI Change: Extend Frequency from 31 days **to** 92 days

(NOTE: Future Exelon T.S. revision request will pursue relocation of STI from T.S. to TRM, and STI extension)

A. SYSTEM & MAINTENANCE RULE (MRule) INFORMATION

SYSTEM NUMBER: 076C

Current MRule R-S Classification: HSS (HSS or LSS)
Current MRule R-S Basis: The SSC is modeled by the PSA. Although the system was not quanitatively risk significant, it was assessed by the Expert Panel to be risk significant.

Current PRA Model:	LGS101r1 and LGS201r1	
Current PRA R-S Classification (System)	non risk significant	
Current PRA RAW (System):	1.0	(MRULE R-S threshold: 2.0)
Current PRA RRW (System)	1.0	(MRULE R-S threshold: 1.005)
Current PRA Limiting Cutset (System)	n/a	(MRULE R-S threshold: top 90%)
NEI 00-04 R-S Insights	Level 2 impact only, no impact on CDF. M. Rule HSS classification retained as allowed by RITS 5b methodology document..	

B. QUALITATIVE ANALYSIS:

1 COMMITMENT REVIEW (Is STI credited in any commitments?)

The following were reviewed to determine if any commitments were impacted as a result of the proposed Technical Specifications change: the UFSAR, General Design Criteria (GDC 43), Limerick Generating Station Commitment Tracking Database, SER's, and Regulatory Guides. The specific basis for the 31 day surveillance interval is not given in the LGS TS Bases section nor in the LGS UFSAR Sections 6.5.1 or 9.4.2 which discuss the subject systems. However, Regulatory Position C.4.d of Regulatory Guide 1.52, Revision 2, relating to maintenance requirements, recommends:

> "Each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the adsorbers and HEPA filters."

The bases for Surveillance Requirements (SR) 3.6.4.3.1 in the Standard Technical Specifications for General Electric Plants BWR/4, which corresponds to the subject LGS TS test, also notes the need for ten (10) hours of operation per month for elimination of moisture in the filters. The basis for the requirement of a monthly test with the heaters energized is clearly related to the desired elimination of moisture in the filters and adsorbers. However, LGS UFSAR Table 6.5-2 states that LGS does not conform to R.G. 1.52, Position C.4.d because the Standby Gas Treatment System (SGTS) and Reactor Enclosure Recirculation System (RERS) trains are "continuously purged with dry instrumentation air to prevent build-up of moisture." UFSAR sections 6.5.1.1.2 and 6.5.1.3.2 provide additional discussion of this method of moisture control.

2 SURVEILLANCE TEST HISTORY OF THE COMPONENTS AND SYSTEM ASSOCIATED WITH THE STI EXTENSION:

The surveillance test, maintenance rule, and corrective maintenance history were all reviewed in answering this section. The results of ST-6-076-250-1(2), SGTS and RERS Flow Test were reviewed for the past five years. All tests were reviewed with the focus on those graded as either "Unsat" or "Fail." There was a total of only four (4) failures of the Unit 1 monthly surveillance test and only one failure of the Unit 2 test in the five year period. It should be noted that all of these test failures were a result of problems on the RERS system only. No failures were caused by SGTS. The review revealed that several other tests were aborted as a result of other issues, for example, high RWCU room temperatures because the normal HVAC system was not in service. These test abortions were not caused by failure of the RERS or SGTS system, and are therefore not counted as "failed" tests.

The Maintenance Rule functional failures were also reviewed for SGTS and RERS for the past seven years. Only five (5) of the fourteen (14) total functional failures were discovered while performing surveillance test ST-6-076-250-1(2), SGTS and RERS Flow Test. Each of these five discovered failures occurred on the RERS system, and not on SGTS. One of these RERS issues resulted in an LER (report number 1-02-004). The other failures were discovered during Operations rounds, during performance of the SGTS Heater DP test ST-4-076-101-0, or while placing the SGTS and RERS in service to support maintenance on the Reactor Enclosure HVAC system.

Although there have been many Corrective Maintenance (CM's) tasks written against the SGTS and RERS systems, only a small number of these issues either resulted in failure of the surveillance test or resulted in the loss of the maintenance rule system or component functionality. Additionally, the majority of these CM's were identified by means other than the monthly surveillance test.

3 RELIABILITY REVIEW:
PERFORMANCE (OPERATION & MAINTENANCE) HISTORY OF THE COMPONENTS AND SYSTEM ASSOCIATED WITH THE STI EXTENSION:

There is no specific reliability tracking mechanism nor any Maintenance Rule performance criteria for reliability associated with SGTS and RERS. The performance indicators (PIs) for SGTS and RERS consist of tracking Limiting Conditions of Operation (LCO) hours and Maintenance Preventable Functional Failures (MPFF). See section B-4 for discussion of PI for LCO hours. The Maintenance Rule functional failures were reviewed for SGTS and RERS. There were only fourteen (14) functional failures on the system in the past seven years; ten were RERS failures and four were SGTS failures. A review was also performed for MPFF's on the system. There was only one MPFF which occurred on SGTS (2001) and only one MPFF which occurred on RERS (2002) in the past seven years. In the past seven years, system 76C reached MRule (a)(1) status only a single time, which was a result of Agastat relay and love controller issues on the SGTS system.

Additional PIMS component history review:

A review of maintenance work requests for the previous five years was performed. A significant number of corrective maintenance (CM) work requests were written to address Love controller failures which caused SGTS or RERS to fail. To address this concern a Love controller replacement project, which is being tracked by the Maintenance Rule, is currently being implemented to replace the existing controllers with new more reliable controller. Another significant contributor to SGTS and RERS problems in the past five years has been the failure of Agastat relays. An Agastat relay program is currently in place to address issues such as end of life (EOL) and preventative maintenance adequacy which will help to improve the reliability of the system. A number of corrective maintenance requests have been written to address high vibration conditions on the SGTS and RERS fans. The development of vibration problems can usually be attributed to wear on system components. Unnecessary wear on these system component could be reduced by a change in the surveillance test frequency. The SGTS components, for example, are common to both units and must be run with the associated RERS system for the surveillance test for each unit. The currently specified test frequency results in the SGTS being run at least twice per month for short durations. A change in surveillance test frequency would reduce the wear on system components and thereby reduce the associated downtime for maintenance and repairs.

4	UNAVAILABILITY REVIEW: MRule Train Actual Unavailability: See below MRule Unavailability Performance Criteria: 7% SGTS, 4% RERS (see below) The unavailability of the SGTS and RERS systems is currently tracked in terms of LCO hours which is conservative compared to unavailability. The performance indicator measures LCO hours averaged over a 24 month period. Unavailability Performance Criteria (limit) for SGTS : 7 % Actual 'A' SGTS Train Unavailability: 1.62 % Actual 'B' SGTS Train Unavailability: 2.08 % Unavailability Performance Criteria (limit) for RERS: 4 % Actual '1A' RERS Train Unavailability: 1.26 % Actual '1B' RERS Train Unavailability: 0.24 % Actual '2A' RERS Train Unavailability: 0.59 % Actual '2B' RERS Train Unavailability: 0.75 %
5	PAST INDUSTRY AND PLANT-SPECIFIC EXPERIENCE WITH THE FUNCTIONS AFFECTED BY THE PROPOSED CHANGES Regulatory Guide 1.52 Rev. 2, position C.4.d recommends monthly operation of the ESF atmosphere cleanup systems to protect the filtration systems. Like Limerick, other nuclear plants get the benefit of detecting additional failures by running SGTS and RERS monthly. However, the detection of failures at other plants is also not limited to performance of the monthly surveillance test. Although many failures were found on SGTS and RERS at various nuclear plants, only a relatively small number of these failures (i.e heater failures) affected the moisture elimination design features. No failures were found on components associated with instrument air injection to the filters.
6	VENDOR-SPECIFIED MAINTENANCE FREQUENCY Vendor manual M-054-C-00028, " CENTRIFUGAL FANS, MAINTENANCE MANUAL" *does not specify a specific time* interval for performing maintenance on the Buffalo Forge centrifugal fans. The RERS were manufactured by JOY fans. Vendor Manual M-069-C-00028 for JOY SERIES 1000/2000 fans does not specify any particular preventative maintenance frequency for the RERS fans. Vibration monitoring is performed by the Predictive Maintenance team for both the RERS and SGTS fans. Currently, this data is taken monthly when the SGTS and RERS fans are run per surveillance test ST-6-076-250-1(2). Preventative maintenance tasks associated with the RERS/ SGTS dampers, instrument loops, electric heaters, and filters are not affected by the performance of this test on a quarterly frequency.
7	ASME AND OTHER CODE-SPECIFIED TEST INTERVAL ANSI N510 " Testing of Nuclear Air Treatment Systems" describes airflow test but does not specify any frequency.

8	OTHER QUALITATIVE CONSIDERATIONS (ex. LCO REVIEW) The instrument air supply used to reduce the buildup of moisture to the charcoal filters on SGTS and RERS has demonstrated high reliability. Safety related solenoid valves are used to provide dry instrument air to the upstream side of the charcoal adsorbers. Whenever the filter systems are not in use, the associated solenoid valve is energized to allow a continuous purge of the adsorbers. Limerick's instrument air is tested to ensure high quality on a regular basis per the requirements of ANSI/ISA S-7.3 and as specified in the PECO response to Generic letter No. 88-14 dated February 13, 1989. The requirements specified in the Improved Technical Specification (ITS) were also reviewed for comparison to current and proposed LGS surveillance requirements. The ITS requires that, every 31 days, each SGTS system be run for more than 15 minutes with heaters operating to eliminate moisture on the adsorbers and HEPA filters. The 31 day frequency was developed in consideration the reliability of fan motors and controls and the redundancy available in the system . ITS also takes credit for the instrument air injection to the filters, but uses it to justify a ≥ 15 minute run per month versus the 10 hours recommended in R.G. 1.52. Note that all plants do not have a Reactor Enclosure Recirculation System (RERS). Note: LGS submitted a LAR in 1996 to change the surveillance interval for SGTS and RERS flow tests from 31 d to 92 d. This LAR was rescinded in 1998. The prior LAR was technically justified; however, PECO withdrew it to enable higher priority (preferred) LGS LARs to be evaluated by the NRC. The basis for the rescinded LAR was reviewed and used to support this RITS STI evaluation.
9	IMPACT ON DEFENSE-IN-DEPTH PROTECTION. There are 2 trains of SGTS shared by Unit 1 and 2. There are 2 trains per unit of RERS. The extension of the STI, given that reliability remains the same does not affect the defense in depth of the plants.
10	THE IMPACT OF SYSTEMS NOT QUANTIFIED USING THE INTERNAL EVENT PRA SGTS and RERS do not affect level 1 PRA.
11	THE IMPACT OF SYSTEMS FOR WHICH LERF RESULTS ARE NOT AVAILABLE The only system not modeled that could impact the importance of SGTS and RERS is additional venting pathways. Some of the paths include for example the ILRT fill line discharge outside of the reactor building and thereby releasing directly to the environment. SGTS and RERS are not involved in this scenario and the reactor building integrity is preserved. Any primary containment failure not involving venting would still fall under the modeled portion of the internal events level 2 PRA.
12	THE IMPACT OF SYSTEMS FOR WHICH EXTERNAL EVENTS AND SHUTDOWN PRA ARE NOT AVAILABLE See section C for quantitative PRA analysis.
13	UNCERTAINTY ASSOCIATED WITH THE QUANTITATIVE (PRA) PROCESS There is little uncertainty associated with this analysis beyond that inherent in the PRA model. The systems are being failed with no impact which is a bounding case in Section C..

14 QUALITATIVE ANALYSIS – CONCLUSIONS

There is no LGS Technical Specification Basis for running the SGTS or RERS systems on a monthly interval. The basis for the monthly surveillance tests comes from the recommendation in Regulatory Guide 1.52 Revision 2, to operate the systems monthly preventing moisture buildup on the filters. Limerick took exception to this recommendation as discussed in LGS UFSAR Table 6.5-2. The design of the SGTS and RERS systems at Limerick includes a safety related feature that injects dry instrument air upstream of the adsorbers when these systems are not operating to prevent build-up of moisture. Therefore a change in the interval between tests from monthly to quarterly will not result in moisture accumulation that would reduce the capability of the adsorbers to remove iodine from the exhaust air flow stream.

Although there have been a number corrective maintenance activities for the SGTS and RERS systems, only a small number of these issues has resulted in a loss of the maintenance rule system function. Additionally, the majority of these CM's were identified by means other than the monthly surveillance test ST-6-076-250-1(2). A change in the surveillance test frequency from monthly to quarterly would be beneficial in that it would reduce the wear on system components and thereby reduce downtime associated with maintenance and repairs. Surveillance test frequency on a quarterly interval is considered adequate to verify operability, as demonstrated by the required quarterly test interval for other equipment important to safety which have a similar function, such as the requirement for quarterly verification of the isolation time of the secondary containment and refueling area isolation valves, as required by LGS T.S. sections 4.6.5.2.1 and 4.6.5.2.2.

15 PHASED IMPLEMENTATION RECOMMENDATIONS

Based on the surveillance test, maintenance rule, and availability history, it is recommended that surveillance test ST-6-076-250-1(2) be changed directly to a quarterly (92 day) frequency.

16 PROPOSED SURROGATE MONITORING RECOMMENDATIONS: (Consider use of Existing MRule monitoring)

The existing monitoring established by the LGS Maintenance Rule is adequate for monitoring performance of the SGTS and RERS systems. Thus, LCO hours (unavailability) and MPFF's will continue to be tracked at the train level.

17 REVIEWERS (Section B – Qualitative Analysis – Reviewers):

Prepared by: ___Allan Charles________________________ (Subject Matter Expert) Date: _11/03/03________
(System Manger or Component Specialist)

Prepared by: __Victoria Warren__________________ (PRA input) Date: _11/06/03___________
(Risk Management Engineer)

C.	PRA (QUANTITATIVE) ANALYSIS	☐ check if not modeled in PRA
1	OVERVIEW OF PRA MODELLING of STI (Include bounding risk analysis techniques if used, and PRA Quality Issues) The potential for SGTS/RERS to provide a mechanism to reduce fission product releases in the Level 2 evaluation is accounted for in the RB (Reactor Building Effectiveness) node of the Level 2 model. However, currently only passive fission product removal mechanisms (such as gravitational settling of aerosols) is credited in providing a limited amount of reactor building effectiveness in the Level 2 evaluation. This is based on the following: 1) For any large containment failure modes, the SGTS/RERS system would be incapable of preventing overpressure failure of the blowout panels in the reactor building. Thus, SGTS/RERS would be bypassed for all large containment failure mechanisms. 2) For small containment failure modes, SGTS could potentially avert overpressure and failure of the blowout panels, but given severe accident scenario conditions, closure of the SGTS/RERS dampers is likely to occur on high gas temperatures such that SGTS/RERS is also currently not credited for reducing the source term released from the Reactor Building in these scenarios. The gap analysis of the LGS PRA against the ASME standard (RA-S-2002) as modified by DG-1122 was reviewed and no identified issues affect this analysis.	
2	FULL POWER INTERNAL EVENTS (FPIE) LEVEL 1 PRA MODEL IMPACTS (CDF Comparison against R.G 1.174 limits) SGTS and RERS are not used in the level 1 PRA because they are release mitigation systems not core damage mitigation systems. As such they have not impact on the Level 1 PRA.	
3	FPIE LEVEL 2 PRA MODEL IMPACTS (LERF Comparison against R.G 1.174 limits) Per the above description in C.1, SGTS and RERS as currently credited and modeled have no impact on LERF for internal events. Calculated delta LERF is 0. A delta LERF of 0 is within the 1.174 limits.	
4	FIRE RISK IMPACTS (CDF & LERF Comparison against R.G 1.174 limits) There is no impact on CDF for fire risk for the same reason as internal events, Similarly there is 0 impact on Fire LERF. The limits of RG 1.174 are therefore met.	
5	SEISMIC RISK IMPACTS (CDF & LERF Comparison against R.G 1.174 limits) There is no impact on CDF for seismic risk for the same reason as internal events, Similarly there is 0 impact on seismic LERF. The limits of RG 1.174 are therefore met.	

6	SHUTDOWN RISK IMPACTS (CDF & LERF Comparison against R.G 1.174 limits) SGTS is typically required for filtering and exhausting of the primary and/or secondary containment atmosphere during refueling or maintenance outages that require opening of the reactor pressure boundary. The assessment of SGTS under shutdown conditions has many of the same attributes associated with the similar analysis performed for the at-power condition. For example, SGTS cannot prevent core damage under power or shutdown conditions. SGTS provides a part of secondary containment. Under shutdown conditions, SGTS provides a method to maintain a negative pressure inside the Reactor Enclosure. The SGTS can filter secondary containment atmosphere before release to the environment. It is capable of radionuclide filtration for events such as a fuel assembly drop accident. These accidents are not sufficient to produce a LERF whether SGTS is operating or not. Therefore, SGTS can be considered to be of low safety significance. The role of the standby gas treatment system in shutdown conditions is primarily the mitigation of radionuclide releases following a Design Basis fuel drop accident. In this accident, gap release from the dropped fuel is released into the Reactor Enclosure. However, this release is extremely small when compared with the releases associated with a postulated full core accident involving prolonged loss of cooling and eventual failure of fuel in the core that could cause a LERF. Although the Standby Gas Treatment System does provide some mitigation of the releases associated with the fuel drop accident, this mitigation potential is not considered safety significant as measured by CDF or LERF due to the relatively small radionuclide release and the potential for success of other protective measures (e.g., evacuation of onsite personnel).
7	OTHER PRA ISSUES (ex. Impacts from Other External Events excluding Seismic & Fire Risk Impacts) None
8	CUMMULATIVE EFFECT OF ALL RI-TS STI EXTENSIONS ON INTERNAL, EXTERNAL & SHUTDOWN PRAs. (CDF & LERF Comparison against R.G 1.174 limits) This is the first quantitative STI extension thus the cumulative impact is the same as the impact for this change
9	QUANTITATIVE (PRA) ANALYSIS – CONCLUSIONS The quantitative PRA analysis supports the extension of the subject surveillance interval to quarterly from monthly.
10	PREPARER (SECTION C – PRA (QUANTITATIVE) ANALYSIS) Prepared by: __Victoria Warren__________ Date _11/06/03______ (Risk Management Engineer)

D	INTEGRATED DECISION-MAKING PANEL (IDP, a/k/a EXPERT PANEL) REVIEW	MEETING DATE:
1	Presenter(s): ;	
2	Meeting Discussion: (Review of Qualitative and Quantitative Analyses, and Cumulative Impact)	
3	Meeting Results / Recommendations / Bases: (Consider: phased implementation, additional performance monitoring of failure rates)	
4	Approval / Disapproval: Check one of the following: ☐ STI Approved ☐ STI Approved with Comments ☐ STI Disapproved **IDP / Expert Panel Members:** 1. Engineering Manager * ________ 2. Maintenance Manager * ________ 3. Operations Manager * ________ 4. Risk Management (PRA) Engineer * ________ 5. Maintenance Rule Coordinator * ________ 6. Surveillance Test Coordinator ________ 7. System Manager or Component Engineer ________ Listing of IDP attendees: (signatures not required – see MRule Expert Panel / IDP meeting minutes) * also Maintenance Rule Expert Panel Member	
5	IDP / Expert Panel Coordinator Final Review / Closure: ____________ Date: ________ (IDP Coordinator)	