May 12, 2011

MEMORANDUM TO: Charles E. Ader, Director
Division of Safety Systems and Risk Assessment
Office of New Reactors

FROM: Donald A. Dube, Senior Technical Advisor **/RA/**
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Office of New Reactors

SUBJECT: SUMMARY OF PUBLIC MEETING TO PERFORM TABLETOP EXERCISES REGARDING GUIDANCE ON RISK-INFORMED INSERVICE INSPECTION OF PIPING FOR NEW REACTORS HELD ON MAY 4, 2011

On May 4, 2011, a public meeting was held at the Two White Flint North Building, Room 2B3, to conduct tabletop exercises regarding the adequacy of existing guidance on risk-informed inservice inspection of piping when applied to new reactor designs. These exercises were performed to address the Commission's Staff Requirements Memorandum (SRM) of March 2, 2011 on SECY-10-0121. The agenda and ground rules, along with preliminary planning material for the second set of workshops, are provided as Enclosure 1. A list of attendees is provided as Enclosure 2. Additional presentation materials prepared by the staff are included in Enclosure 3. Handouts presented by industry representatives are provided as Enclosures 4 and 5.

The workshop was the first in a series in response to the Commission SRM to perform tabletop exercises that "test various realistic performance deficiencies, events, modifications, and licensing bases changes against current U. S. Nuclear Regulatory Commission policy, regulations, guidance and all other requirements (e.g., Technical Specifications, license conditions, code requirements) that are or will be relevant to the licensing bases of new reactors."

The Electric Power Research Institute (EPRI) provided an overview of its methodologies on risk-informed inservice inspection of piping. Most risk analysts would categorize the impact of changes to the inservice inspection program as being risk-neutral. The staff provided sample results from actual licensee submittals for currently operating reactors supporting the point that the theoretically calculated changes in core damage frequency (CDF) and large early release frequency (LERF) are sometimes positive, sometimes negative, but virtually always low in absolute magnitude.

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EPRI provided scoping calculations of the potential impact of a RI-ISI program for a new active plant and a new passive plant that showed the effects continued to be risk-neutral, even when sensitivity studies using more restrictive acceptance criteria were assumed.

That should not be surprising, in that RI-ISI simply changes the locations for inservice inspection in a risk-informed manner. Neither the design nor plant operational configurations are affected by RI-ISI.

Meeting participants identified the regulatory and programmatic controls that would tend to limit the decrease in enhanced safety margin of the new reactor design. These include, for example:

- The guidelines on potential increases to the baseline CDF and LERF are imposed at a system level as well as the overall totals. This ensures that no one system absorbs most of the change in risk
- Inspection of a minimum set of weld locations is required regardless of what the risk levels are calculated to be
- A number of programs remain in place to address degradation mechanisms such as flow accelerated corrosion and microbiologically induced corrosion
- Risk category 4 in the risk evaluation matrix was introduced in the EPRI methodology to address the unknowns with high consequence/low frequency phenomena
- Risk category 5 was introduced to ensure that some inspection is provided even if the consequences of certain pipe failures are identified as low
- The RI-ISI program requires updating the risk ranking, on average, every 3 and 1/3 years; this interval approximates the Part 52 requirement for periodic upgrade of the plant-specific PRA
- The 10-yr ISI program is dynamic and allows for incorporation of lessons-learned

The general consensus from the tabletop is that use of current guidance for RI-ISI for a new reactor design with sufficient operating experience would not result in any significant decrease in enhanced safety. There are some potential regulatory and implementation issues that would need to be addressed if RI-ISI was applied to a new plant lacking operating experience. Presently, a conventional ISI program per 50.55a is a requirement to implement RI-ISI. However, these are not “risk-metric” issues.

At the end of the workshop, participants discussed plans including scope and schedule for the next series of tabletop exercises. Two days were deemed necessary to accommodate key participants for risk-informed technical specifications (RITS) initiative 4b, along with Maintenance Rule 50.65(a)(4).

The revised schedule for the next few rounds of tabletop exercises are as follows:

1) May 26 and June 1, 2011: RITS initiative 4b, along with 50.65(a)(4)
2) June 29, 2011: RITS initiative 5b, surveillance frequency program
3) August 9 or 10, 2011: 50.69, risk-informed categorization of structures, systems, and components (SSCs), including both passive and active SSCs.

In June and July, 2011, NRO staff will review the new Appendix C to NEI 96-07, "Guideline for Implementation of Change Control Processes for New Nuclear Power Plants Licensed under 10 CFR Part 52." Among other things, this document will address change control processes that may impact ex-vessel severe accident features in new reactor designs.

Enclosures:
As stated

In June and July, 2011, NRO staff will review the new Appendix C to NEI 96-07, "Guideline for Implementation of Change Control Processes for New Nuclear Power Plants Licensed under 10 CFR Part 52." Among other things, this document will address change control processes that may impact ex-vessel severe accident features in new reactor designs.

Enclosures:
As stated

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Public Workshop to Perform
Tabletop Exercises Regarding Existing Guidance on
Risk-Informed Inservice Inspection of Piping for
New Reactors

8:00 a.m. to 5:00 p.m.
Wednesday, May 4, 2011
NRC, Two White Flint North, Rockville, MD (Room T-2B3)

TIME	TOPIC	LEAD
8:00 – 8:10 a.m.	Introduction and opening remarks	NRC
8:10 – 8:30 a.m.	Ground rules for workshop	NRC
8:30 – 9:30 a.m.	Overview of EPRI methodology on risk-informed inservice inspection of piping	EPRI
9:30 – 9:45 a.m.	Break	NRC
9:45 – 10:30 a.m.	Discussions of risk impacts of RI-ISI for operating reactors	NRC, Industry
10:30 – Noon	Qualitative and quantitative discussion of results and projected risk impacts for new reactor designs	Industry
Noon – 1:00 p.m.	Lunch	
1:00 – 1:45 p.m.	Continued discussion of results for new reactors	All
1:45 – 2:15 p.m.	Sensitivity study results using more restrictive risk metrics	EPRI
2:15 – 3:15 p.m.	Discussion of regulatory and licensee controls to limit the decrease in enhanced safety margin for new reactors	All
3:15 – 3:30 p.m.	Break	
3:30 – 4:45 p.m.	Discussion and preliminary conclusions regarding adequacy of existing guidance	All
4:45 – 5:00 p.m.	Major lessons learned for next workshop	All
5:00 p.m.	Adjourn	

Workshop #1, SRM to SECY-10-0121: Tabletop on Risk-Informed Inservice Inspection of Piping	
Date	May 4, 2011
Location	NRC, Rockville, MD, room T2B3 (ACRS sub-committee room)
Time	8 am to 5 pm
Objective of workshop	To test licensing basis changes related to RI-ISI for new reactor designs, and either confirm the adequacy of existing regulatory guidance or identify areas for improvement
Scope of Workshop	Limited to issues of the adequacy of the existing risk-informed guidance. While legitimate programmatic and process issues related to implementation of RI-ISI at new reactors may arise, any such issues will be set aside and will be addressed outside this workshop.
Regulatory guidance	RG 1.178, RG 1.174
Supporting document(s)	Revised Risk-Informed Inservice Inspection Evaluation Procedure (PWRM RP-05), TR-112657 Revision B-A, Final Report, December 1999 (ML013470102)
New reactor designs in tabletop	One reactor design with active safety features and one with passive features
SPAR model(s)	None to be used for this workshop
Further Commission direction per SRM	"If the staff concludes that the enhanced safety margins for new plants will significantly decrease without regulatory policy changes, the staff should clearly explain how 'significant' (in the context of decreasing safety margins) was defined to support the recommendations."
Pre-workshop activities	1. Industry and NRC staff to review a sample of past licensing basis change submittals on RI-ISI and to categorize the "theoretical" impact of the change in terms of risk neutral, risk decrease, or risk increase. 2. Industry (EPRI and its contractors) to qualitatively, and to the extent possible, quantitatively, assess the risk impact of implementing the EPRI methodology on (preferably) 2 new reactor designs a. Assuming non-RI-ISI as "baseline" b. Assess sensitivity of results to alternate "risk metrics"
Workshop activities	1. Brief overview of EPRI RI-ISI methodology, including code case(s) to be used (e.g., N-716) 2. Qualitative and quantitative discussions of risk-impacts of RI-ISI for currently operating reactors (see item #1 on pre-workshop activities) 3. Qualitative and quantitative discussions of risk-impacts of RI-ISI for at least two new reactor designs (preferably one with active safety features, one with passive) 4. Identification of a) regulatory controls, and b) licensee controls to limit the decrease in the enhanced safety margin for new reactors
Preliminary conclusion to draw from tabletop exercise	Determine whether the preponderance of the experience on RI-ISI from the currently operating fleet, qualitative and quantitative results of the tabletop exercises, and the regulatory and licensee controls to limit the decrease in the enhanced safety margin a) provide reasonable assurance of the adequacy of existing risk-informed guidance when applied to new reactor designs, or b) identify the need for additional analysis or tabletop exercises, and if so, what additional analysis/tabletop, what time frame, and the owner(s) of such action item, or c) whether an area for improvement has been identified, the technical basis for concluding a "significant" decrease in the enhanced safety margin will result, and the specific recommendation to be made to the Commission
Lessons-learned	A list of the major lessons learned from the workshop/tabletop should be carried forward to future workshops/tabletops.

Workshop #2, SRM to SECY-10-0121: Tabletop on Configuration Control, RITS 4b and 50.65(a)(4)	
Date	TBD (most likely date is May 26, 2011 with back-up dates May 25 or June 1)
Location	TBD (either NRC headquarters or nearby hotel)
Time	8 am to 5 pm
Objective of workshop	To test configuration control processes associated with implementation of Risk-Informed Technical Specifications (RITS) initiative 4b and Maintenance Rule 50.65(a)(4) for new reactor designs, and either confirm the adequacy of existing regulatory guidance or identify areas for improvement
Scope of Workshop	Limited to issues of the adequacy of the existing risk-informed guidance to prevent significant decrease in the enhanced margin of safety for new plants. Process issues will not be addressed in this workshop.
Regulatory guidance	RG 1.177, RG 1.174, RG 1.182
Supporting document(s)	1. NEI 06-09, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines 2. NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, draft Rev. 4, Section 11 3. AP1000 Design Control Document (DCD), Section 16.3.1, Investment Protection Short-Term Availability Controls 4. ESBWR DCD, Section 19ACM, Availability Controls Manual and Bases
New reactor designs in tabletop	At least one reactor design with active safety features and one with passive features
SPAR models	AP1000 and ABWR
Further Commission direction per SRM	"If the staff concludes that the enhanced safety margins for new plants will significantly decrease without regulatory policy changes, the staff should clearly explain how 'significant' (in the context of decreasing safety margins) was defined to support the recommendations."
Pre-workshop activities	1. Industry to review experience with RITS 4b at STP 1&2 and identify scenarios to tabletop for new designs 2. Industry to assess experience with implementation of 50.65(a)(4) including scenarios to tabletop for new reactor designs 3. Qualitative and quantitative discussions of risk-impacts of RITS 4b and 50.65(a)(4) for at least two new reactor designs. See template to report results. 4. NRC staff to use SPAR models to augment risk assessment of various scenarios of equipment outages
Workshop activities	1. Discussion of experience with RITS 4b at STP 1&2 2. Qualitative and quantitative discussions of risk-impacts of RITS 4b and 50.65(a)(4) for at least two new reactor designs; on-line configuration control demo. 3. Identification of a) regulatory controls, and b) licensee controls to limit the decrease in the enhanced safety margin for new reactors
Preliminary conclusion to draw from tabletop exercise	Determine whether the preponderance of the experience at STP Units 1&2 on RITS 4b, the overall industry experience on 50.65(a)(4) for the currently operating fleet, qualitative and quantitative results of the tabletop exercises, and the regulatory and licensee controls to limit the decrease in the enhanced safety margin a) provide reasonable assurance of the adequacy of existing risk-informed guidance when applied to configuration control processes such as RITS 4b and 50.65(a)(4) for new reactor designs, <u>or</u> b) identify the need for additional analysis or tabletop exercises, and if so, what additional analysis/tabletop, what time frame, and the owner(s) of such action item, <u>or</u> c) whether an area for improvement has been identified, the technical basis for concluding a "significant" decrease in the enhanced safety margin will result, and the specific recommendation to be made to the Commission
Lessons-learned	A list of the major lessons learned from the workshop/tabletop should be carried forward to future workshops/tabletops

template for Workshop #2								
RITS 4b Case	**Equip. Not Functional**	**CDF* (per yr)**	**ΔCDF (per yr)**	**Calc Completion Time (days)**	**Tech Spec Limit (days)**	**Allowed Completion Time (days)**	**ICDP**	**Other Available Equip**
Base	None (no T&M)	2.6E-07	--	--	--	--	--	All
1	1 EDG	later	later	later	14	later	later	2 offsite AC power sources, 2 EDGs, and CTG
6	RCIC	later	later	later	14	later	later	2 HPCFs and 3 LPFLs
13	1 EDG and RCIC	later	later	later	EDG - 14 RCIC - 14	later	later	2 offsite AC power sources, 2 EDGs, and CTG; 2 HPCFs and 3 LPFLs

* total CDF for all events and modes if available; only internal events at-power is available for ABWR SPAR model

EDG	emergency diesel generator
CTG	combustion turbine generator
RCIC	reactor core isolation cooling
HPCF	high pressure core flooder
LPFL	low pressure flooder