RULEMAKING ISSUE
INFORMATION

January 30, 2009 SECY-09-0018

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: STREAMLINING DESIGN CERTIFICATION RULEMAKINGS

PURPOSE:

To inform the Commission of staff actions that are being implemented in order to streamline the design certification rule (DCR) rulemaking process for new reactor designs. This paper does not address new commitments.

BACKGROUND:

Recently the nuclear industry has shown significant interest in licensing new reactors. The Nuclear Regulatory Commission (NRC) has docketed one combined license (COL) application incorporating by reference the U.S. Advanced Boiling-Water Reactor, a previously certified design (Appendix A, "Design Certification Rule for the U.S. Advanced Boiling-Water Reactor," to 10 CFR Part 52). Currently, 10 COL applications docketed by the NRC incorporate by reference designs that have been recently submitted to the NRC for certification, such as the U.S. Evolutionary Power Reactor, the U.S. Advanced Pressurized-Water Reactor, and the Economic Simplified Boiling-Water Reactor. In addition, six COL applications reference an amendment to the Advanced Passive 1000 (AP1000) design (Appendix D, "Design Certification Rule for the AP1000," to 10 CFR Part 52). As a result, the staff is reviewing COL applications in parallel with the NRC review of the design certification (DC) applications being referenced.

The referenced DCR must be completed (i.e., the final rule published in the *Federal Register*) before the NRC can make a decision on the COL application referencing that DC. The DC schedule consists of: (1) the design review and issuance of a final safety evaluation report documenting the NRC's safety conclusions related to the design; and (2) a rulemaking approved by the Commission that codifies that DC in the agency's regulations. The review schedules for

CONTACT: George M. Tartal, NRO/DNRL
(301) 415-0016

certain DCs and their reference COLs are projected to be nearly parallel, with the DCR being completed just before the completion of the COL hearing process and subsequent licensing decision. If the DCR is not issued by the time that the NRC is ready to make a decision on the COL application, the COL decision may, under certain circumstances, have to await the issuance of the final DCR.[1] Based on this background the staff concluded that a rulemaking schedule shorter than the typical 24-month NRC rulemaking schedule would be of value for DCR rulemakings.

DISCUSSION:

The staff recognized the potential for the DCR to become part of the critical path for a COL application decision and proceeded to look for schedule improvements to the DCR process. The Office of the Executive Director for Operations (OEDO) approved a request by the Office of New Reactors (NRO) to conduct a Lean Six Sigma (LSS) review of the DCR process. A team of NRC subject matter experts (the team) joined with an OEDO LSS black-belt trainee and the OEDO LSS contractor to undertake a Kaizen (rapid improvement) event. The overall goal of the Kaizen event was to identify improvements that would aid completion of the final rule before the scheduled date for a decision on the COL application. Before conducting the review, the team identified two subgoals in achieving the overall goal — (1) reduce the overall duration of the rulemaking process to 12 months or less; and (2) start the rulemaking earlier in the DC process.

During the Kaizen event, the team first thoroughly examined the rulemaking process by breaking down the proposed and final rule phases into several steps and substeps. Time frames for each of the steps and substeps were estimated based on the team's experience with previous rulemakings, including those of the four currently certified designs. The team's evaluation shows that, when applying the current rulemaking process, completion of a DCR could take 19-23 months (this paper subsequently uses 19.5 months as a best estimate). This conforms to the expectation for an average NRC rulemaking (1 year for the proposed rule and 1 year for the final rule). The team then identified several staff-initiated process changes that could be implemented to streamline the rulemaking process specifically for DCRs. The staff is currently implementing those changes, which do not involve policy issues The team, initially, did not include stakeholders, such as representatives from the Advisory Committee on Reactor Safeguards (ACRS), during the Kaizen event. Upon further consideration, the team discussed its proposal to streamline the DCR rulemaking process with the ACRS and considered other potential opportunities as described in the enclosure. The staff-initiated process changes being implemented and other changes considered but not being pursued by the staff are described further in the enclosure. The durations of the nine rulemaking process phases before and after implementing the process changes are also described.

Each staff-initiated process change is intended to enhance the NRC's ability to promulgate DCRs efficiently and effectively and contribute to the NRC's successful execution of its Strategic Plan. More specifically, these process changes will aid the NRC in meeting the Strategic Plan organizational excellence objectives of openness, effectiveness, and operational excellence. The openness goal is met by maintaining an adequate public comment period for the proposed

[1] The COL applicant could voluntarily amend its application in a manner in which the plant's design would be regarded as a "custom" design and would not necessitate a rulemaking. This would be a significant change to the application and is unlikely to be considered a practical alternative to the DCR process.

rule and using a standard communications plan to promptly inform internal and external stakeholders of the NRC's rulemaking activities. The effectiveness goal is met by ensuring that DCRs are published within the scheduled COL time frame while still meeting all NRC policy and legal requirements. The operational excellence goal is met by optimizing business processes (e.g., concurrence and intra-agency reviews) and using standardized document templates and procedures to streamline the rulemaking process.

The staff-initiated process changes will reduce the DCR schedule by approximately 7 months (from 19.5 months to 12.5 months). Meeting the subgoal of an earlier start of rulemaking activities complements other process changes in meeting the overall project goal. Therefore, as a result of these process changes, the NRC should be able to meet the overall goal of coordinating the DCR and COL schedules such that the final rule is completed to support a decision on the first COL application referencing each DC application.

RESOURCES:

Resources for the planned process changes are included in the Fiscal Year (FY) 2009 budget. NRO has requested 2.2 full-time equivalents in its FY 2010 budget to work on NRO's highest priority rulemakings. Resources for FY 2011 and beyond will be requested through the Planning, Budget, and Performance Management process.

COORDINATION:

The staff has discussed its proposed focused scope of review as described in the enclosure with ACRS. This paper has been coordinated with the Office of General Counsel (OGC), the Office of Nuclear Reactor Regulation and the Office of Administration (ADM). OGC has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this Commission paper for resource implications and has no objections. The NRC's Rulemaking Coordinating Committee has been informed of those identified process changes that could be applied generically to other rulemakings.

OGC and ADM will review the document templates, procedures, and other products as they are developed to ensure that the DCR rulemaking process continues to conform to the NRC's rulemaking policies and procedures and all applicable statutes and regulations.

/RA/

R. W. Borchardt
Executive Director
for Operations

Enclosure:
LSS Methodological Approach to Determine
Actions Needed to Streamline DCRs

LEAN SIX SIGMA METHODOLOGICAL APPROACH TO DETERMINE ACTIONS NEEDED TO STREAMLINE DESIGN CERTIFICATION RULEMAKINGS

Approach

Lean Six Sigma (LSS) is a highly structured methodology used to accomplish sustained improvements to processes, transactions, and services. It combines two improvement approaches: making work faster (using Lean principles) and making work better (using Six Sigma). Following LSS methodology, a team, comprised of subject matter experts and process stakeholders, defines the processes, identifies opportunities for improvements, and brainstorms potential solutions. LSS requires close cooperation and communications among team members and their representative stakeholders to maximize the likelihood that: (i) the ultimate decision maker will approve of the LSS process improvement recommendations, and (ii) the LSS process improvements will be successfully implemented. LSS involves systematic identification of *all* potential opportunities for reduction in time and resources necessary to achieve the desired product – in this LSS project, to issue a final design certification (DC) rule in time to support, without delay, a decision on a referencing COL.

LSS eliminates "non-value added" steps thereby reducing time and resources needed to achieve the project goals. In LSS terminology, "non-value added" refers to activities that add no value from the customer's perspective and are not required for legal, financial, or other business reasons. Examples of non-value added activities include overproduction, over processing, rework, duplicative work, and waiting/idle time. In the context of DC rulemaking, non-value added activities also include process steps which are not necessary to meet the needs of the COL applicant referencing (or considering referencing) the DC.[1]

Design Certification Rulemaking Kaizen Event

Due to the small amount of historical data and the need for improvement, the team decided to perform its LSS review using the LSS Kaizen (rapid improvement) event methodology. A Kaizen event is a focused, intense, short-term event to improve a process within the scope of the process participants. It usually includes training followed by process analysis, solution brainstorming, and implementation design. A Kaizen event normally takes 5 days and the results are intended to be immediate, dramatic and satisfying. The overall goal of the design certification rule (DCR) streamlining Kaizen event was to identify improvements that would ensure completion of the final rule before the scheduled date for a decision on the COL application. Before conducting the review, the team identified two subgoals in achieving the overall goal—(1) reduce the overall duration of the rulemaking process to 12 months or less and (2) start the rulemaking earlier in the DC process.

During the Kaizen event, the team first thoroughly examined the rulemaking process by breaking down the proposed and final rule phases into several steps and substeps. Time frames for each of the steps and substeps were estimated based on the team's experience with previous rulemakings, including those of the four currently certified designs. The team's evaluation showed that, when applying the current rulemaking process, completion of a DCR

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1 Although elimination of non-value added activities could also benefit a design certification applicant, the staff believes that the design certification was not an end in itself but was adopted by the Commission as a more effective regulatory approach for licensing nuclear power plants.

could take 19-23 months (this paper subsequently uses 19.5 months[2] as a best estimate). This conforms to the expectation for an average U.S. Nuclear Regulatory Commission (NRC or the Commission) rulemaking (1 year for the proposed rule and 1 year for the final rule). The team then identified several staff-initiated process changes that could be implemented to streamline the rulemaking process specifically for DCRs. The staff is currently implementing those changes, which do not involve policy issues. The team, initially, did not include stakeholders such as representatives from the Advisory Committee on Reactor Safeguards (ACRS), during the Kaizen event. Upon further consideration, the team discussed its proposal to streamline the DCR rulemaking process with the ACRS and considered other potential opportunities. The staff-initiated process changes being implemented and other changes considered are described below.

Staff-Initiated Design Certification Rulemaking Process Changes

1. *Dedicate a rulemaking project manager (PM) from the Office of New Reactors (NRO) to each DC rulemaking*

In past DCRs one of the PMs for that DC was assigned to manage the review of the design control document (DCD) and issue the final safety evaluation report (FSER), as well as manage the DCR. To manage each of these projects (FSER and DCR) more efficiently and effectively, this process change will assign a separate and dedicated PM to work under the direction of NRO management and manage one DCR. The term "dedicated," as used in this context, means that the DCR is the highest priority activity for that PM. Only during periods in the DCR schedule when the rulemaking PM is not actively working on the DCR would that individual be available to work on other tasks. This change provides its greatest benefit during the package preparation phases of the rulemaking process.

2. *Develop standard document templates, procedures, and training*

As discussed in the body of this paper, by the time the NRC issued the fourth existing DC, the regulatory text and Statement of Considerations had become largely standardized. This process change will improve upon that concept by developing standardized DCR-specific templates for all documents required for, and to support concurrence on, the rulemaking package. These templates will highlight those parts of the documents that are design specific, thus enabling the PM and other reviewers to focus on those areas and ensure standardization for all future DCRs. Procedures will be developed to guide the PMs through each phase of the rulemaking processes and focus other concurring/interfacing offices on specific parts of the package that are more relevant to that office. Training will be developed for and provided to the PMs on how to use the templates and procedures. This change provides its greatest benefit during the package preparation and concurrence phases of the rulemaking process.

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[2] This includes a total of 3.0 months for Commission review of the proposed and final rules based on historical data of Commission review times for the previous four DCRs. This is discussed further under "Other Streamlining Considerations" later in this enclosure.

3. *Start the rulemaking when the advanced FSER for the design is under review by the ACRS*

Under the current DC project schedule, DCRs are scheduled to start 3 months before the FSER is issued. This process change will start the DCR when the ACRS begins its review of the advanced FSER, currently 2 to 5 months earlier than under the current schedule. The staff's review of the design is nearly complete at this phase of the FSER development, and significant FSER or design changes are not expected. However, as the design review progresses, this earlier start time continues to change and varies between DC schedules. While this change provides no schedule reduction benefit, it provides additional margin between the final rule publication and the COL issuance dates, thus supporting the second subgoal of this effort.

4. *Optimize the concurrence process*

Under the current concurrence process, all of the branch-, division-, and office-level reviews are performed sequentially. That is, branches complete their review before the divisions, and the divisions complete their reviews before the offices. Several entities at each level of management may review the package in parallel, but not in parallel with different levels of management. This process change includes a number of actions to facilitate timely concurrence on DCRs. A DCR steering committee will be established to focus management attention on the allocation of resources and the resolution of issues that may impact concurrence on that rulemaking(s) (see process change number 5). All branch, division, and office concurrences will be performed in parallel to facilitate the concurrence process. In a concurrence meeting, all concurring branches, divisions, and offices will meet to discuss and resolve comments and to provide their concurrence at the end of the meeting. The staff will also eliminate any unnecessary concurrences from offices that might typically be asked to concur on rulemakings (e.g., the Office of Enforcement), but only with prior agreement of those offices. The concurrence of the Executive Director for Operations (EDO) will be completed within 2 weeks. This change provides benefits during the concurrence phases of the rulemaking process.

5. *Initiate a working group and steering committee*

Working groups are currently used on many NRC rulemakings and provide the benefits of involving key staff members in the rulemaking and helping to streamline concurrence through the staff's branch, division, and/or office management. Under this process change, NRO will establish a working group for each DCR that is composed of staff from key concurring offices. Working group members will help identify and resolve any issues that could delay preparation of or concurrence on the DCR package. The working group will present issues that it cannot resolve to a steering committee, which will be responsible for timely and independently resolving any interoffice or technical/policy issues. The steering committee will be comprised of division-level management from NRO and other NRC offices as appropriate, and may be the same committee that oversees similar issues for DCs and COLs. This change provides benefits during the concurrence and comment resolution phases of the rulemaking process.

6. *Manage the impact of the information collection approval process*

Currently, the NRC policy is to withhold the publication of the final rule until after receiving the Office of Management and Budget's (OMB's) approval of the information collection

requirements contained in the final rule. Within 60 days OMB may approve, instruct the NRC to make a substantive change to, or disapprove, the collection of information contained in the final rule. The staff will examine options for streamlining OMB approval of information collections contained in DCRs with the goal of minimizing the potential for delay in the issuance of the final rule. The staff will review the criteria for what constitutes an insignificant information collection burden and determine if these rulemakings qualify for processing as an insignificant change in burden. The staff will also evaluate whether it is possible to obtain a generic approval for DCRs, with the understanding that the minimal increases in burden will be reflected in the information collection budget. This change may provide, in certain cases, benefits during the OMB clearance phase of the rulemaking process.

7. *Inform applicants of the consequences of late design changes*

The vendors for each of the four currently certified designs made changes to those designs during the rulemaking phase of DC. As a result, the staff was required to review the changes and provide a supplement to the issued FSER. With the currently submitted license applications referencing designs under review for certification, late design changes could cause significant delays in the rulemaking schedule and adversely affect the NRC's ability to support the reference COL, and possibly subsequent COL, schedules. The staff will discuss the consequences of late design changes by a DC applicant with each of the new reactor design centers. The staff will determine the DC schedule milestone that would constitute a "late" change. The staff then plans to write a letter to the DC applicants to formally inform them of the NRC staff's position on late design changes and their consequences. While this change may not provide a schedule reduction benefit, it reduces the risk that any late design changes would necessitate a DCR schedule revision.

8. *Proposed rule need not reference the FSER as a published NUREG*

In past DC rulemakings, the staff has converted the FSER document into a published NUREG before the proposed rule's publication. The Office of General Counsel (OGC) has determined that a DCR proposed rule can be published without the FSER being converted into and published as a NUREG. Instead, under this process change, the FSER can be placed into the Agencywide Documents Access and Management System (ADAMS) and made publicly available with no reduction in the public availability of that information. During the final rule phase, the FSER will be converted to a NUREG and the DCR reference to the FSER will be revised to reflect the NUREG number assigned to it. In its initial assessment, the team assumed that the ADAMS version would be sufficient. However, without it, the DCR process for publishing the proposed rule could be delayed by approximately 2 months.

9. *Management review of changes to staff requirements memorandum (SRM)*

Management should be informed of any SRM requirements imposed on DCRs. However, the team decided that management could be informed using informal communications (e-mail, meetings, briefings, etc.), rather than requiring a formal reconcurrence on the DCR. The staff will resolve and incorporate all comments from the Commission and use informal communications to inform management of SRM-related changes to the final rule. The team assumed this change when estimating schedule durations. However, without it, the concurrence process during both the proposed and final rule phases could be delayed by approximately 1–2 weeks.

10. *Optimize ACRS review time*

The provisions of 10 CFR 52.53 require ACRS to report to the Commission on its safety review of each DC application. The team believed that ACRS fulfills this requirement during its review of the staff's FSER and the DC application (including the DCD), and that ACRS need not separately review the proposed or final DCR. The team noted that ACRS waived its review of the Advanced Passive 600 (AP600) and Advanced Passive 1000 (AP1000) final rules.

The team decided that optimizing the review of DCRs by ACRS was appropriate. Because of the standardized rule language for DCRs, and because ACRS has previously reviewed the technical basis (DC application and associated FSER) for the rulemaking, the team decided that the ACRS review could be optimized by focusing on technical comments made on the proposed rule. As a result, the staff proposed to send all technical comments on the rule and the staff's resolution of those comments to ACRS for its review instead of the entire rulemaking package. Furthermore, because ACRS briefings would not be expected for these focused scope reviews, they would not be scheduled but provided upon request. Because the ACRS waived its review of the AP600 and AP1000 final rules, the team assumed ACRS briefings would not be needed on subsequent DCR final rules. If a briefing is requested, the DCR process for publishing the final rule could be delayed by 0.5 to 2 months, depending on the timing of the fixed ACRS briefing dates with the concurrence process on the final rule.

During the 556th meeting of ACRS, October 2–4, 2008, the team discussed with ACRS its proposal to optimize ACRS review of DCRs. The ACRS expressed concern with the team's proposal because, should ACRS agree to this proposal and subsequently request a staff briefing, they would be viewed as forcing their way back into the process and thus be responsible for delaying the DCR. The ACRS suggested the staff use the normal process for requesting a waiver of ACRS' review of a rulemaking. The ACRS would consider the individual proposal, expedite its decision, and not further delay the DCR process. The ACRS recommended that the staff not change this part of the rulemaking process for DCRs, but rather plan for an ACRS review and briefing as part of the generic DCR schedule. If ACRS agrees with the staff's proposal, it could waive its review. However, this should be considered on a case-by-case basis. Based on this input, the staff will not propose a change at this time, but will continue to evaluate its process and procedures for seeking ACRS review of DCRs, with the goal of optimizing the ACRS review process.

Table 1. Time Savings (Months) Resulting from Staff-Implemented Process Changes

Process Change	Time Savings
Dedicate an NRO rulemaking PM to each DCR	1.5
Develop standard document templates, procedures, and training	2.75
Start the rulemaking when the design FSER is under review by the ACRS	0[3]
Optimize the concurrence process	1
Initiate a working group and steering committee	0.25
Manage the impact of the information collection approval process	1
Inform applicants of the consequences of late design changes	0[4]
Proposed rule need not reference the FSER as a published NUREG	0[4]
Management review of SRM changes	0.5
Optimize ACRS review time	0[4]
TOTAL SAVINGS	7.0

Schedule Improvements

As a result of the 10 improvements described above and in Table 1, the staff estimates that the DCR rulemaking process can be shortened from 19.5 months to 12.5 months. The results of streamlining each phase of the DCR rulemaking process are shown in Figure 1 and Table 2 below.

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3 Although this change shows no net time savings, it contributes to the subgoal of starting the rulemaking earlier in the DC process to minimize the impact on COL schedules. The amount of contribution varies between designs because each DC schedule is unique and changes as the design review progresses.

4 The team assumed these changes in its initial assessment of rulemaking process durations. Although these changes show no net time savings, they could further extend the rulemaking process if they are not adhered to or are not implemented.

Figure 1. DCR Rulemaking Schedules Before and After Streamlining

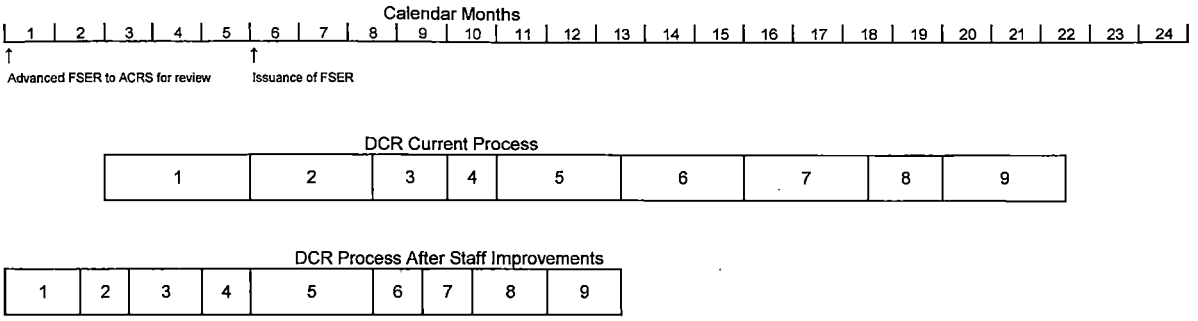

Table 2. DCR Rulemaking Phase Durations (Months) Before and After Streamlining

Phase	Description	Current Duration	New Duration
1	Prepare proposed rule	3.0	1.5
2	Proposed rule concurrence	2.5	1.0
3	Proposed rule Commission review / SRM	1.5	1.5
4	Publish proposed rule	1.0	1.0
5	Public comment period	2.5	2.5
6	Prepare final rule	2.5	1.0
7	Final rule concurrence	2.5	1.0
8	Final rule Commission review / SRM	1.5	1.5
9	OMB & Office of the Federal Register approvals; Publish final rule	2.5	1.5
	TOTALS	19.5	12.5

Other Streamlining Considerations

The team considered two other possible opportunities for streamlining the DCR rulemaking process: 1) Reducing the public comment period to 60 days, and 2) Streamlining the Commission's review and approval. As described below, when weighing the potential benefits against their drawbacks, the staff has decided to not further consider either opportunity.

1. Reduce the public comment period to 60 days

The team considered whether the 75-day comment period could be reduced, consistent with legal requirements. Neither the Administrative Procedure Act nor the Atomic Energy Act of 1954, as amended, requires a specific amount of time for a public comment period on rulemaking. The NRC staff's current practice of either recommending (in a rulemaking that the Commission acts on) or using (in a rulemaking where the Commission has delegated rulemaking authority to the EDO) a 75-day comment period is derived from Executive Order (EO) 12889, "Implementation of the North American Free Trade Agreement" (NAFTA), dated December 27, 1993. For each of the four existing DCRs, the staff determined that 75 days should be the minimum duration provided for submission of public comments, consistent with EO 12889 and NAFTA. For the first two DCRs (the U.S. Advanced Boiling-Water Reactor

(ABWR) and the System 80+), however, the NRC provided a 120-day public comment period; supplemental proposed rules were published for 30-day comment periods with 60-day extensions granted. For the third and fourth DCRs (the AP600 and the AP1000), the NRC provided a 75-day public comment period. Although the NRC received more comment letters for the first two DCRs, which resulted in the need to resolve policy issues for DCRs, the NRC received only one comment letter during the AP600 comment period and four comment letters during the AP1000 comment period. In light of the small number of stakeholder comments in the last two DCRs, the standardized nature of the rule language and the DCD form and content, and the Commission's authority to grant an extension to the public comment period upon request of an external stakeholder, the staff considered the use of a 60-day public comment period for a future proposed DCR. During its consideration of this process change, the staff determined that a reduction in the public comment period could be negatively perceived as emphasizing the licensing schedule over public participation in the DC rulemaking. In addition, if a shorter public comment period were provided, it is possible that a member of the public would request an extension of the comment period such that the comment period would exceed the 75-day period that the NRC would have ordinarily provided. If granted, the extension of the public comment period would likely exceed 15 days and negate any benefit this change could have provided. That is, the schedule risk (i.e., receiving and granting an extension of the public comment period) in implementing this change far outweighs the small benefit it would have provided. Therefore, the staff did not recommend implementing a reduced public comment period.

2. *Optimize Commission review time*

During the Kaizen event, the team identified possible ways to optimize the Commission review time and voting on DCRs. The team considered the possible schedule improvements if the Commission were to delegate to the EDO the authority to issue the proposed rule for public comment, the final rule following resolution of public comments, or both the proposed and final rules. Of these three options, the team considered the delegation of the proposed rule to be the most viable option given the standardized nature of DCR rulemakings, the sharing of information with the Commission regarding the staff's review of each design, and the Commission's role as the ultimate decision-making body in the final rulemaking. However, following discussions with the team and with OGC and NRO, the staff determined that such delegation raises a number of concerns, regardless if such delegation is done generically or on a case-by-case basis. These concerns include: (i) the fairness of the adjudicatory process for COL applications referencing DC applications; (ii) the possibility that some external stakeholders may have reduced confidence in the NRC's regulatory process where DC applications are referenced by COL applications – regardless of whether there is an adjudicatory hearing on the referencing COL application; and (iii) the perceived (if not actual) reduction in effective Commission oversight over DC rulemakings. The staff has decided that no changes to current process are necessary.

Conclusion

The staff-initiated process changes will reduce the DCR schedule by approximately 7 months (from 19.5 months to 12.5 months). Although the modified schedule falls short by 0.5 month of the subgoal of this project to complete the rulemaking in 12 months (or as quickly as possible), meeting the subgoal of an earlier start of rulemaking activities complements other process changes in meeting the overall project goal. Therefore, as a result of these process changes,

the NRC should be able to meet the overall goal of coordinating the DCR and COL schedules such that the final rule is completed to support a decision on the first COL application referencing each DC application.

The staff will use existing briefings and updates to inform the Commission about the various aspects of new reactor licensing, including schedules and relationships between specific DCRs and their related licensing proceedings. In order to minimize schedule issues and ensure that information is provided to the Commission in a timely manner and decisions are not unnecessarily deferred to the rulemaking process, the staff will continue its practices from previous DCRs by:

(1) Seeking early Commission direction on policy and regulatory issues during the design review and inform the Commission of significant technical issues contained in the original DC application and responses to requests for additional information.[5]

(2) Providing an information copy of each design review advanced FSER to the Commission when it is provided to the ACRS.

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5 It would be difficult to seek Commission guidance in advance of the proposed DCR package on technical or regulatory issues that result from amendments to the DCR application filed near the end of the staff's technical review. The staff also notes that previous DCR applicants have filed amendments to their DCD *after* the close of the public comment period on the proposed DCR, but that the staff, in consultation with OGC, determined that those DCD changes did not require renoticing of the DCR in the *Federal Register* for public comment.

POLICY ISSUE
(Information)

December 2, 2009 SECY-09-0174

FOR: The Commissioners

FROM: Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

SUBJECT: STAFF PROGRESS IN EVALUATION OF BURIED PIPING AT NUCLEAR REACTOR FACILITIES

PURPOSE

This paper responds to the Chairman's memorandum dated September 3, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092460648) tasking the staff to describe the activities currently underway or planned addressing the issue of leaks from buried piping. In his memorandum, Chairman Jaczko requested a staff evaluation of the adequacy of (1) Nuclear Regulatory Commission (NRC) requirements for design, inspection and maintenance of safety-related buried piping; (2) American Society of Mechanical Engineers Code (ASME Code) requirements for design, inspection, and maintenance of safety-related piping; and (3) voluntary initiatives for the design, inspection, and maintenance of safety-related and nonsafety-related buried piping. The Chairman also requested a discussion of staff plans for recommending any revisions to regulations, requirements, practices or oversight related to buried piping.

BACKGROUND

Over the past several years, instances of buried piping leaks have occurred in safety-related and nonsafety-related piping at nuclear power plants. Some of these leaks have caused inadvertent releases of low-level radioactive material and diesel fuel oil. This has resulted in groundwater contamination at several plants. The pipe degradation leading to these leaks has not affected the operability of safety systems, and the type and amount of radioactive material or chemicals released to the environment have been a small fraction of the regulatory limits. Consequently, these pipe leaks have been of low significance with respect to public health and safety.

CONTACT: Robert O. Hardies, NRR/DCI
301-415-5802

The NRC staff has evaluated these recent leakage events, root cause evaluations and licensee corrective actions. Based on application of NRC's performance-based and risk-informed Reactor Oversight Process (ROP), these recent events did not constitute performance deficiencies of greater than minor safety significance because the radiological consequences associated with the leaks were very low and because the capability of the piping to perform its safety function was not degraded. Upon application of NRC's Issue Screening and Significance Determination Processes, the staff determined the issues were of minor safety significance.[1] Accordingly, these buried pipe leakage events have not warranted enforcement actions.

In May 2006, the U.S. commercial nuclear power plants adopted the Nuclear Energy Institute (NEI) Groundwater Protection Initiative (GPI) in response to leaks containing radioactive material at several plants. The initiative is described in Attachment 1 to "NEI-07-07 Industry Ground Water Protection Initiative - Final Guidance Document," dated August 2007 (ADAMS Accession No. ML062260198). The initiative, which all plants committed to follow, identifies actions to improve licensee response to inadvertent releases that may result in low but detectible levels of plant-related radioactive materials in subsurface soils and water. The GPI provides the actions licensees are expected to take including the development of written groundwater protection programs, improved stakeholder communications, and program oversight. One objective of the GPI is to detect leaks well before they can challenge regulatory limits for unintended release of radioactive material. The GPI addresses detection and remediation of leaks but is not focused on preventing leaks.

NUREG/CR-6876, "Risk-Informed Assessment of Degraded Buried Piping Systems in Nuclear Power Plants," evaluated corrosion damage on buried pipe and concluded that structural integrity can be maintained even with relatively high levels of general wall thinning. Because corrosion of buried piping typically initiates at local areas of coating damage, the resulting degradation is localized. From a structural integrity perspective, localized corrosion is much less challenging than general wall thinning. Localized degradation of buried piping typically causes small leaks that do not challenge structural integrity. Reports of operating experience are consistent with this observation; there have been no challenges to functionality or structural integrity due to degradation of buried, safety-related piping at nuclear power plants.

DISCUSSION

Review of Current Regulations

The NRC staff reviewed applicable regulations governing buried, safety-related piping for operating reactors, renewal of licenses, and new reactor licenses. For all nuclear power plants, the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria," or similar requirements imposed during plant licensing for pre-10 CFR Part 50 Appendix A facilities, and 10 CFR 50.55a, "Codes and Standards," provide design requirements for safety-related components, including some buried piping. In addition, 10 CFR 50.55a provides requirements for examining and testing buried, safety-related piping.

[1] Note that leakage at Braidwood Station in 2005, described in NRC Information Notice 2006-13, "Ground-water Contamination Due to Undetected Leakage of Radioactive Water" was caused by a leaking vacuum breaker and not by degradation of buried piping; therefore, it was not within the scope of this review.

Licensees are required to verify that radioactive effluents, either from pipe leakage or from normal operations, are within NRC regulatory limits and design objectives. The NRC limits for radioactive effluents are contained in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and the design objectives are contained in 10 CFR 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." These limits and design objectives are incorporated into licensee's technical specifications and Offsite Dose Calculation Manual.

For new plants where license applications are submitted after August 20, 1997, 10 CFR 20.1406 "Minimization of Contamination" requires a description of how the facility's design and procedures for operation will minimize (to the extent practicable) contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste.

For plants applying to renew their licenses, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," establishes requirements for managing aging effects of certain systems and components, including buried piping, that are important to safety or whose failure could adversely impact the ability of systems to perform their intended safety function.

With regard to buried piping, the goals of current regulations are to ensure that the piping is able to perform its intended safety function by supplying sufficient fluid flow and to maintain inadvertent releases below licensee's technical specifications or other applicable limits. The staff has determined that current regulations are adequate for these purposes.

Review of the ASME Code

The staff reviewed applicable portions of the ASME Code. The ASME Code, Section III, "Rules for Construction of Nuclear Facility Components," provides design rules for materials, design, fabrication, installation, examination, testing and overpressure protection to ensure the structural integrity of nuclear piping and components. The ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," provides requirements for the examination and testing of buried, safety-related piping. Section XI also provides acceptance criteria, rules for the evaluation of flaws, and repair/replacement rules for piping and components. The ASME Code does not currently address nonsafety-related piping nor does it address leaks that are not structurally significant.

The staff has determined that current ASME Code requirements are adequate for ensuring structural integrity is maintained for buried, safety-related piping such that the piping will be capable of performing its safety function. The staff routinely works with ASME to update the Code to address operational experience or improvements in design or inspection technology and will continue this participation.

Inspection

The staff reviewed NRC inspection guidance on licensee activities related to buried piping degradation. Under the ROP, the baseline inspection program allows inspectors to review issues associated with buried piping if they are believed to be risk significant. There is no direct guidance to select a buried piping issue as an inspection sample; however, inspectors can

review flow tests, inservice inspection activities, condition monitoring, post-maintenance testing, and corrective actions using select inspection procedures. Historically, most buried pipe ROP inspection samples have been selected as a followup to an actual piping failure, known degradation, or problem identified during a review of a condition report. In addition, NRC reviews licensee implementation of the industry Groundwater Protection Initiative.

The staff has determined that the current inspection process is adequate to verify appropriate licensee implementation of current regulations and the GPI. The staff plans to evaluate and revise, as necessary, the ROP to address new industry initiatives for buried pipe.

Industry Activities

The staff has engaged industry, including at public meetings on August 20, 2009, and October 22, 2009, to gather information on the scope and status of their activities related to buried piping. Specifically, the staff met with representatives from a specific licensee with recent buried piping leakage experience, the Institute of Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI), and NEI.

INPO conducts performance-based evaluations at each plant about every two years. In 2007 INPO identified buried piping as a focus area and began to assess buried piping degradation management using predictive maintenance criteria.

In December 2008, EPRI published "Recommendations for an Effective Program to Control the Degradation of Buried Pipe," to provide nuclear power plant licensees with guidance on implementing preventive maintenance programs to detect and mitigate degradation in piping systems before leakage occurs. The industry has recently begun to implement this guidance.

By letter dated November 20, 2009 (ADAMS Accession No. ML093350032), NEI indicated that the nuclear industry's chief nuclear officers voted to approve a proposed "Buried Piping Integrity Initiative." The stated goal of the initiative is to "provide reasonable assurance of structural and leakage integrity of all buried piping with special emphasis on piping that contains radioactive materials." Objectives include proactive assessment and management of the condition of buried piping systems and technology development to improve upon available techniques for inspecting and analyzing underground piping. The staff plans to meet with the industry to further understand this initiative and assess licensee implementation.

Conclusions

Based on the staff's review of operating experience related to buried piping degradation, current regulations and ASME Code requirements have been effective in ensuring that the structural integrity and functionality of buried, safety-related piping are maintained. Current regulations have also been effective in ensuring unintended releases of hazardous material to the environment from leaks in buried piping remain below regulatory limits. Therefore, the staff has no current plans to recommend regulatory changes to address degradation of buried piping.

The industry has recently developed the Buried Piping Integrity Initiative. The staff plans to meet with the industry to further understand this initiative and monitor industry implementation. The staff will also evaluate the need to revise NRC Inspection Procedures to assess licensee implementation of this new initiative.

The staff will continue to monitor operating experience and assess the need for any further regulatory actions or communications. The enclosure to this paper provides additional detail related to these conclusions.

COORDINATION

The Office of the General Counsel has reviewed this paper and has no legal objection.

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosure:
Additional Information Related to
Evaluation of Buried Piping Degradation
at Nuclear Reactor Facilities

ADDITIONAL INFORMATION RELATED TO EVALUATION OF BURIED PIPING DEGRADATION AT NUCLEAR POWER PLANTS

PURPOSE

This paper provides additional detail related to the Chairman's memorandum dated September 3, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092460648) tasking the staff to describe the activities currently underway or planned addressing the issue of leaks from buried piping.

INTRODUCTION

In his memorandum, Chairman Jaczko requested a staff evaluation of the adequacy of (1) Nuclear Regulatory Commission (NRC) requirements for design, inspection and maintenance of safety-related buried piping; (2) American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements for design, inspection, and maintenance of safety-related piping; and (3) voluntary initiatives for the design, inspection, and maintenance of safety-related and nonsafety-related buried piping. The Chairman also requested a discussion of staff plans for recommending any revisions to regulations, requirements, practices or oversight related to buried piping.

The staff responded to the Chairman tasking memorandum with SECY-09-0174, Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities (ADAMS Accession No. ML093160004). This paper provides additional detail related to conclusions made in SECY-09-0174.

BACKGROUND

Over the past decade, instances of buried piping leaks have occurred in safety-related and nonsafety-related piping at nuclear power plants. Some of these leaks have caused inadvertent releases of low-level radioactive material. This has resulted in groundwater contamination at several plants. The staff evaluated these events individually and contemporaneously. The pipe degradation leading to these leaks did not affect the operability of safety systems, and the type and amount of radioactive material or chemicals released to the environment was a small fraction of the regulatory limits. Consequently, these pipe leaks have been of low significance with respect to public health and safety and the environment.

In June 2005, as part of a program to assess a variety of forms of age-related degradation in passive components, the staff published NUREG/CR-6876, "Risk-Informed Assessment of Degraded Buried Piping Systems in Nuclear Power Plants." This report evaluated corrosion damage on buried pipe and concluded that structural integrity can be maintained even with relatively high levels of general wall thinning.

ENCLOSURE

In May 2006, the U.S. commercial nuclear power plants adopted the Nuclear Energy Institute (NEI) Groundwater Protection Initiative (GPI) in response to abnormal releases of radioactive material at several plants (which were not necessarily related to buried piping degradation). The initiative is described in Attachment 1 to "NEI-07-07 Industry Ground Water Protection

Initiative - Final Guidance Document," dated August 2007 (ADAMS Accession No. ML062260198). The initiative, which all plants committed to follow, identifies actions to improve licensee response to inadvertent releases that may result in low but detectable levels of plant-related radioactive materials in subsurface soils and water. These actions include the development of written groundwater protection programs, improved stakeholder communications, and program oversight. One objective of the GPI is to detect leaks well before they can challenge regulatory limits for unintended release of radioactive material to the public and the environment. The GPI addresses detection and remediation of leaks but is not focused on preventing leaks.

Degradation of buried piping is caused by corrosion. Structural integrity of buried piping can be readily maintained because corrosion normally occurs at localized points associated with coating damage. This corrosion has little effect on the structural integrity of the pipe because little or no corrosion occurs in areas where coating damage does not occur. Instead, localized degradation of buried piping typically causes small leaks that do not challenge structural integrity. Reports of operating experience are consistent with this observation; there have been no challenges to functionality or structural integrity due to degradation of buried, safety-related piping at nuclear power plants. Safety-related piping that has been excavated for the purposes of repairing leaks has been inspected to determine whether degradation has affected the structural integrity of the piping. In no cases has structural integrity been compromised.

EVALUATION

In preparing its response to the Chairman's tasking memorandum, the staff evaluated operational experience, regulations, codes and standards, NRC inspection activities, industry activities and international activities. The following sections discuss the staff's evaluation, conclusions and follow-up actions for each of these areas.

Operational Experience

The staff operating experience clearinghouse performed a screening of several events involving leakage from buried piping and on July 16, 2009 identified and assigned buried piping as an Issue for Resolution (IFR). As part of the IFR generic evaluation process, the staff collected and evaluated recent operational events that were reported to the NRC. The NRC staff has evaluated these recent leakage events, root cause evaluations and licensee corrective actions. Based on application of NRC's performance-based and risk-informed Reactor Oversight Process (ROP), these recent events did not constitute performance deficiencies of greater than low safety significance because the radiological consequences associated with the leaks were very minor and because the piping remained capable of performing its safety function. Upon

application of NRC's Issue Screening and Significance Determination Processes, the staff determined the issues were of low safety significance.[2]

The staff performed a review of a database of operational events that is maintained by the Institute of Nuclear Power Operations (INPO) and includes descriptions of events that are not required to be reported to the NRC. The review did not reveal any additional instances (not already reported to the NRC) of degradation of safety-related buried piping or inadvertent release of radioactive material. The staff also reviewed NRC inspection findings related to buried piping. None of the findings involved events where piping functionality or structural integrity was compromised.

Conclusions

Based on a review of operational events, the staff concluded that the pipe leaks from these events have been of low significance with respect to public health and safety and the environment.

Actions

The staff is scheduled to issue the IFR report during the second quarter of FY2010 and will take any additional action, if necessary, based on the results of this generic evaluation.

The staff will continue to review operating experience of buried pipe as part of its normal operating experience review process.

Review of Current Regulations

The NRC staff reviewed applicable regulations governing buried piping. These regulations establish criteria or limits that, if met, ensure the health and safety of the public are maintained. The criteria and limits for safety-related piping require the piping to be able to perform its safety function and require that any radioactive material that may be released not pose any credible threat of harm to public health and safety. The staff evaluated whether current regulations establish appropriate criteria with respect to degradation of and release from buried piping.

The review encompassed regulations related to safety function and to release of radioactive material. Within these categories of safety function and release of radioactive materials the staff evaluated regulations related to operating plants, license renewal and new plants. Safety-related is defined by Title 10 of the *Code of Federal* Regulations (10 CFR) 50.2 and means those items relied upon to remain functional during and following design basis events to

[2] Note that leakage at Braidwood Station in 2005, described in NRC Information Notice 2006-13, "Ground-water Contamination Due to Undetected Leakage of Radioactive Water" was caused by a leaking vacuum breaker and not by degradation of buried piping; therefore, it was not within the scope of this review.

assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, as applicable. Safety related piping systems generally fall into one of the ASME Code classifications (i.e. Class 1, 2 or 3). Piping Code classification is determined by using Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" and 10 CFR 50.55a. Safety-related buried piping falls under ASME Code Class 3, which means it generally contains relatively low pressure and low temperature water.

For all nuclear power plants, the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria," or similar requirements imposed during plant licensing for pre-10 CFR Part 50, Appendix A facilities, provide requirements for systems, structures, and components important to safety. These regulations specify piping, including buried piping, be designed in a robust manner that ensures piping structural integrity, capability to withstand the effects of natural phenomenon such as earthquakes, configuration to permit inspection to assure integrity and capability of the system, and capability to permit appropriate periodic pressure and functional testing. The criteria presented in 10 CFR 50, Appendix A, are design and operational objectives that provide a foundation for safety at nuclear power plants. The process of degradation of buried piping does not create any challenge to safety that is not already addressed by the existing GDC. Adoption of changes to the GDC to address buried piping degradation would be redundant to existing GDC. For example, a proposed enhancement to the GDC to require buried safety-related piping to be designed to be capable of withstanding natural events would be redundant to the existing requirements that all safety-related piping, including buried piping, be designed to withstand natural events. Accordingly, the staff concludes that the existing GDC are adequate, so do not need to be enhanced to address buried piping degradation.

In addition, the regulations in 10 CFR 50.55a, "Codes and Standards," require the application of various codes and standards such as the ASME Code. Section III of the Code applies to the design of safety-related pressure-boundary components (including buried piping) in nuclear power plants. Section XI of the Code provides requirements for the examination and testing of safety-related buried piping. Among the requirements established by Section XI is a requirement to perform periodic flow tests of safety-related piping, including buried piping. The flow tests ensure piping segments are capable of performing their safety function of delivering fluid in the appropriate quantity upon demand. In practice plants test safety-related piping every ninety days. The trend from the results of this testing indicate buried safety-related piping continues to be able to perform its safety function even in instances where leaks have occurred. The safety function has not been compromised due to degradation. The staff concludes that the requirements in 10 CFR 50.55a to perform testing are adequate and that the results of the testing indicate that buried piping degradation is not currently impacting the ability of safety-related buried piping to perform its safety function.

10 CFR 50.55a requires that once leakage through the wall of Class 3 piping is discovered, it must be repaired. However, the regulation permits the licensee to postpone the repairs for up to 24 months, depending on operational circumstances. These requirements apply equally to

buried piping and to piping that may be more accessible, such as inside buildings. Licensees are required to evaluate the leakage to ensure the piping remains capable of performing its safety function until such time as repairs can be accomplished. These requirements provide flexibility to perform repairs on a schedule that permits adequate planning while still ensuring the piping always remains capable of performing its safety function.

Licensees are required to maintain radioactive effluents, either from pipe leakage or from normal operations, within NRC regulatory limits and design objectives. The NRC limits for radioactive effluents are contained in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and the design objectives are contained in 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As

Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." These limits and design objectives are incorporated into the licensing basis for each site (e.g., licensee's technical specifications and Offsite Dose Calculation Manual (ODCM)). The effluent concentration limits of 10 CFR 20, Appendix B, Table 2 (10 CFR 20.1302(b)) would be applicable to members of the public. These regulations limit the potential dose to members of the public that could result from an inadvertent release of radioactive material.

The staff evaluated whether the occurrences of release of radioactive material into the ground as a result of degradation of buried piping indicated a need to establish limits at a location other than the site boundary (within the site boundary). As part of the operational experience review, the staff determined that none of the events associated with degradation of buried piping resulted in releases that exceeded a small fraction of existing limits for members of the public at the site boundary. A tabulation that describes the radiological consequences of the leakage during the events that occurred in 2009 is provided in Table 1. As part of the industry's GPI, licensees monitor groundwater using wells, to understand whether leakage of radioactive materials is occurring. Once radionuclides are detected in groundwater onsite, licensees are required (by their licensing basis such as Technical Specifications and ODCM) to monitor ground water and drinking water if local supplies are likely to be affected. The intent of this routine monitoring is to assure that leaks will be detected in a timely manner, before established limits could be challenged. Since the staff's review of the known pipe leaks indicated the dose impact was a small fraction of existing limits at the site boundary, the staff concluded that it is not currently necessary to establish any new requirements that would be applicable inside the site boundary to address buried piping degradation.

Leakage from degraded buried piping can create contaminated soil. Worker dose accumulated during any necessary maintenance or remediation activities is controlled by site radiation protection programs and these activities have not exceeded small fractions of regulatory dose limits. Accordingly, the staff concludes that no changes to worker dose limits are warranted as a result of leakage from buried piping.

Tritium is a naturally occurring element. Its presence in public drinking water supplies is limited by Environmental Protection Agency regulations to be at concentrations no greater than 20,000 picocuries per liter (pCi/l). Most of the leaks from buried piping have been of water containing

less than 20,000 pCi/l tritium, including the recent leak at Indian Point Unit 2, where the leaking water contained 2000 pCi/l, approximately one tenth the limit for drinking water. Some of the leakage events (Dresden in June, 2009, for example) involved water contaminated with tritium at concentrations that exceed EPA limits for safe drinking water. However, the water that has leaked from degraded buried piping is not intended for consumption. Additionally, water that leaks from buried piping becomes diluted as it moves away from the location of the leak. The effects of dilution are such that none of the leaks of tritiated water that have occurred have resulted in leakage offsite that exceeds a fraction of EPA limits for drinking water. This discussion is provided to illustrate the relatively low magnitude of tritium leakage from buried piping.

In addition to the above requirements which address operating power plants, 10 CFR Part 54 establishes additional requirements for power plants seeking to renew their operating licenses. These regulations require that the effects of aging be managed for structures, systems, and components which are passive and long lived and which are safety related; are not safety related, but whose failure could adversely affect safety related functions; or are relied upon to demonstrate compliance with certain regulations. License renewal regulations apply to some, but not all buried piping.

The license renewal rule requires applicants for license renewal to demonstrate that for each applicable structure, system, or component, the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation. This is normally accomplished through the development of Aging Management Programs (AMPs). To assist in this process the staff has provided examples of AMPs in the Generic Aging Lessons Learned (GALL) report (NUREG-1801). Applicant programs which are consistent with the GALL AMPs are generally considered to be acceptable methods for managing aging. The GALL report contains several AMPs related to aging of the interior and exterior surfaces of buried piping.

Applicant's AMPs are generally considered adequate to manage aging when they are consistent with the GALL AMPs and when the plant operating experience is no more adverse than the conditions for which the GALL AMP was designed. Several recent license renewal applications have included additional excavations and visual inspections, nondestructive examinations (various forms of ultrasonic testing), and selective replacement of some piping in response to adverse operating experience or recognized increased susceptibility of piping to aging.

The GALL report is currently undergoing a scheduled revision. During this revision all the aging management programs, including those associated with internal and external corrosion of buried piping are being examined and revised as necessary to address emerging issues. Current plans for the revision of these AMPs are to adjust or clarify inspection procedures, frequencies and locations to conform to current industry practices but to retain the goal of preventing loss of function, rather than establishing a goal of leak tightness, for these pipes. The revised GALL report, including revisions to the AMPs related to buried piping, is scheduled to be published in December 2010.

The GALL report is one acceptable way to meet the regulations. However, the staff concludes that no enhancements to regulations are required to address degradation of buried piping for plants applying for renewed licenses.

New reactors will be required to meet the requirements discussed above. In addition, they will need to comply with 10 CFR 20.1406 (Minimization of contamination), which requires license applications submitted after August 1997 to demonstrate how the facility's design and procedures for operation will reduce contamination of the facility and environment and the generation of radioactive waste. Proper implementation of this regulation for new facilities should substantially reduce or eliminate the occurrence of residual contamination for the next generation of nuclear facilities, and could include adoption of predictive maintenance practices to ensure leakage and contamination is maintained as low as reasonably achievable. It should be noted that this rule permits a licensee to address residual radioactivity by either reducing it or by provisioning additional decommissioning funding to address remediation of the residual radioactivity.

The staff has prepared a draft final rule (SECY 09-0042 - March 13, 2009) for consideration by the Commission related to minimization of contamination such as from leaks from buried piping at current operating facilities. Implementation of such a rule change would provide an incentive to operating facilities to institute predictive maintenance programs that minimize leaks from buried piping.

In practice, the combined operating license applications currently under review do not have safety-related buried piping. All of the safety related piping is proposed to be either above ground or installed in vaults or chases that provide accessibility and the ability to capture any leakage. If an applicant submitted a new design certification for consideration that featured buried safety-related piping, NRC staff reviewers would request information from the applicant describing how they intended to satisfy the requirements of 10 CFR 20.1406.

Conclusions

With regard to buried piping, the goals of current regulations are to ensure that the piping is able to perform its intended safety function by supplying sufficient fluid flow and to maintain inadvertent releases below licensee's technical specifications or other applicable limits which apply at the site boundary. The staff has determined that current regulations are adequate for these purposes.

Actions

The revised GALL report, including revisions to the AMPs related to buried piping, is scheduled to be published in December 2010.

Code and Standards

For all nuclear power plants 10 CFR 50.55a endorses the ASME Code. The ASME Code, Section III, "Rules for Construction of Nuclear Facility Components," provides design rules for materials, design, margins, fabrication, installation, examination, testing and overpressure

protection to ensure the structural integrity of safety-related nuclear piping and components. These rules ensure that installed piping, including buried piping, is robust and capable of performing its safety function. Degradation of buried piping is manifested primarily as localized corrosion that is caused, in part, by choice of maintenance practices that do not emphasize prevention of corrosion. Section III rules are applied to buried piping before it is placed into the ground, so do not specify or address maintenance or long term corrosion protection activities. The staff considered recent examples of buried piping degradation and determined that most of the events occurred on nonsafety-related piping, that structural integrity was not challenged in any of the events, and that the degradation generally occurred because of a choice that the licensee made regarding maintenance practices. In order to change Section III so that it addressed the situations observed in the operating events, its scope would need to be expanded to 1) piping systems that it currently does not address (nonsafety related piping is not within the scope of Section III), 2) a purpose that is not currently within its objectives (Section III requires structural integrity rather than leak tightness), and 3) address processes that it does not currently address (Section III addresses design and installation, not maintenance).

Therefore, the staff concludes that current Section III rules are adequate for their intended purpose of specifying a robust piping system and no changes are warranted.
The ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," provides requirements for the examination and testing of buried, safety-related piping. Section XI also provides acceptance criteria, rules for the evaluation of flaws, and repair/replacement rules for piping and components. Periodic flow testing required by Section XI ensures that buried piping remains capable of performing its safety function to deliver a specified quantity of liquid on demand. The staff considered whether such testing, which offers retrospective proof that safety-related buried piping is capable of performing its safety function, provide sufficient information about the future. Section XI only requires the testing be performed every approximately three years, but in practice all 104 nuclear power plants perform flow testing every ninety days. The results of quarterly tests from the start of plant operation indicate that degradation of buried piping has not compromised its ability to perform its safety function. Accordingly, the staff concludes that current ASME Code Section XI requirements are adequate for ensuring structural integrity is maintained for buried, safety-related piping such that the piping will be capable of performing its safety function.

The staff routinely works with ASME to update the Code to address operational experience or improvements in design or inspection technology and will continue this active participation. Examples of current Code activities that demonstrate evolution of the Code in response to operational experience include a committee that has been empanelled to provide a liaison function between Section XI and Section III so that experience from operating plants can be transmitted to and addressed by Section III, with the intent that new plants are able to avoid some challenges that were faced by older plants. Additionally, a Section III committee is developing updated design rules specifically for buried piping. The proposed changes involve incorporating newer soil loading calculative procedures. Once Section III publishes the newer design rules, Section XI intends to implement flaw evaluation procedures specific to buried piping. A final example of current Code activities related to buried piping involves the application of high density polyethylene (HDPE) as a material for buried piping. Buried HDPE piping has been extensively used in the water and gas industries, where its advantages of low cost, ease of fabrication and immunity to corrosion make it an attractive alternative to coated

steel. The ASME is developing material property information related to long-term mechanical performance of this material in order to justify its addition as an acceptable material for application in buried, Class 3 applications. The staff is actively participating in all of these activities.

Recognizing that leaks in buried piping, if undetected for extended periods, could represent precursors to loss of structural integrity, the staff discussed the issue of buried piping degradation informally with Code members during committee meetings in November 2009. ASME Code committee members pointed out that, in general, the industry examples of degradation of buried piping had been discovered in nonsafety-related, nonclass piping, which were not subject to the jurisdiction of the ASME Code. Participants indicated that extension of ASME Code jurisdiction to nonsafety-related piping would be a large undertaking without an obvious benefit in terms of safety. With respect to the infrequent occurrences of degradation of safety-related piping, ASME Code committee personnel noted that the observed degradation did not affect structural integrity. They observed that changes to the Code to require leak tightness would be a significant effort that would not necessarily have any beneficial effect on structural integrity. They concluded that such changes to the Code were probably not warranted.

Codes and Standards in other Industries

Recognizing that many facilities and installations other than nuclear power plants employ buried piping, the staff evaluated regulations and codes and standards applicable to other industries. Buried piping and buried tanks used to contain or transport potentially hazardous or environmentally sensitive material are regulated under Department of Transportation (40 CFR) and Environmental Protection Agency (49 CFR) regulations. Generally, these regulations implement installation, corrosion protection, maintenance and condition monitoring standards developed by NACE International (formerly National Association of Corrosion Engineers). The implementation of these standards has been highly effective in preventing leaks in buried piping and buried tanks. The staff reviewed these standards and concluded that implementation of these standards at nuclear power facilities could be an effective means of reducing the potential for degradation and consequential leakage from buried piping. These corrosion protection standards are required to be applied to many thousands of miles of buried petroleum product transportation piping and many hundreds of thousands of buried tanks in the United States, but are not required to be implemented at nuclear power plants. Current NACE Standards have not been optimized for use by nuclear power plant operators. NACE has recognized the significant interest by nuclear power plant operators in the issue of buried piping and has formed a "Nuclear Buried Piping" task group. The purpose of this task group is to evaluate the need for specific corrosion protection standards that could be implemented at nuclear power facilities. The staff is participating in this committee, which will hold its first meeting during Spring 2010.

Conclusions

While the ASME Code Sections III and XI are adequate to ensure the capability of safety-related buried piping to perform its safety function, implementation of NACE Standards could be an effective means of reducing the potential for degradation and consequential leakage from buried piping.

Actions

The staff will continue to participate in ASME and NACE committees to develop enhancements related to advancements in technology or application of buried piping.

Inspection of Licensee Activities under the Reactor Oversight Program

The staff reviewed NRC inspection guidance on licensee activities related to buried piping degradation. Under the Reactor Oversight Program (ROP), the baseline inspection program does not require specific inspections of licensees' oversight of buried piping. The ROP allows inspectors to review issues associated with buried piping if they are believed to be risk significant. There is no direct guidance to select a buried piping issue as an inspection sample; however, inspectors can review flow tests, inservice inspection activities, condition monitoring, post-maintenance testing, and corrective actions using select inspection procedures. Historically, most buried pipe ROP inspection samples have been selected as a follow-up to an actual piping failure, known degradation, or a problem identified during a review of a condition report.

Examples of buried piping activities inspectors may review are as follows.

1. Inspectors may observe a licensee's periodic flow or pressure test of underground pipes, the video or results of a video or camera inspection of underground piping, results of a sound detection system used to detect leaks in underground piping, or the periodic maintenance conducted on a cathodic protection system (or similar system) used to protect underground piping and detect any potential leaks. These reviews can be captured under IP 71111.19, "Post Maintenance Testing" or IP 71111.22, "Surveillance Testing." Inspectors would review the licensee activities against any facility specific standards or procedures.

2. Inspectors may review a licensee's activities associated with buried piping during an inservice inspection (ISI) as one of the sample requirements in IP 71111.08, "Inservice Inspection Activities," which requires a review of two or three types of Non-Destructive Examination (NDE) activities. The licensee actions would be reviewed against the ASME Code requirements for the ISI activity. Most often for buried pipe this entails a flow test or pressure test.

3. If a buried pipe is within the licensee's maintenance rule scope, inspectors can evaluate the handling of the pipe performance or condition monitoring using IP 71111.12, "Maintenance Effectiveness."

4. If a licensee excavates underground piping for the purpose of repair and replacement, inspectors can use this opportunity for direct visual inspection of the piping. These activities can be reviewed under IP 71111.17, "Evaluations of Changes, Test, or Experiments and Permanent Plant Modifications" or IP 71111.19, "Post Maintenance Testing."

5. When a licensee implements corrective actions due to suspected underground piping leaks (as potentially identified by tritium issues, tank inventory loss, chemical loss, excessive running of pumps on fire protection piping or other "keep pressure" systems, as sink hole, etc.), an inspector can review the licensee's actions under IP 71152, "Identification and Resolution of Problems" or IP 71111.15, "Operability Evaluations."

An article was generated for the July edition of the quarterly Inspector Newsletter. This article provided the background associated with buried piping and detailed how these baseline NRC inspection program tools can be used to evaluate buried piping issues.

License renewal inspections are conducted to support the NRC's review of the license renewal application and to review the licensees programs for managing aging effects on systems, structures and components (SSCs) that fall under the scope of the license renewal. IP 71002, "License Renewal Inspection" is conducted to verify that passive, long lived SSCs within the scope of license renewal are subject to an aging management review and have existing or planned aging management programs (AMPs) that conform to descriptions contained in the license renewal application. The inspection also verifies that AMPs can reasonably manage the effects of aging for these SSCs. IP 71003, "Post-Approval Site Inspection for License Renewal" verifies that a licensee has completed the necessary actions to comply with the license and has implemented the AMPs included in the staff's license renewal safety evaluation report.

Groundwater Protection Initiative (GPI) inspections are performed using Temporary Instruction (TI) 2515/173, "Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative." The objective of this one-time inspection is to assess ground water protection programs to determine whether licensees have implemented the voluntary industry GPI.

As discussed below, the industry has recently instituted a Buried Piping Integrity Initiative. While evaluating the establishment and implementation of this initiative, the staff may perform some audits of selected licensees and may develop a TI to assess the effectiveness of this initiative.

Conclusions

Based on its evaluation of operating experience discussed above, the staff has determined that the priority placed on buried piping degradation within the current inspection process is adequate to verify appropriate licensee implementation of current regulations and the GPI.

Actions

The staff may perform audits and/or develop a TI to evaluate licensee implementation of the Buried Piping Integrity industry initiative. In addition, the staff will continue to use existing inspection tools to evaluate buried piping integrity issues on an as available basis.

Industry Activities

The staff held several public meetings with industry to gather information on the scope and status of their activities related to buried piping. For example, the staff met with INPO at a public meeting on October 22, 2009, to discuss buried piping issues. INPO conducts performance-based evaluations at each plant approximately every two years. These evaluations address the performance of passive components, such as piping and heat exchangers. In 2007, INPO identified underground piping as a focus area and began to assess buried piping degradation management using predictive maintenance criteria described in NACE standards. INPO has completed at least one evaluation of every plant and has elected to continue evaluating buried piping during the current two-year inspection cycle. During the October 22, 2009, meeting industry executives indicated that, in part due to the INPO inspection results, the industry decided to initiate some activities with the Electric Power Research Institute (EPRI) to develop specific guidance for buried piping maintenance.

In December 2008, EPRI published "Recommendations for an Effective Program to Control the Degradation of Buried Pipe'" to provide nuclear power plants with guidance on implementing preventive maintenance programs to detect and mitigate degradation in buried piping systems before leakage occurs. The staff has met with EPRI representatives and attended training on the guidance, which is modeled on the NACE standards. The industry has recently begun to implement this guidance.

In the past, the industry has reacted to operational events by establishing initiatives, such as the Groundwater Initiative (NEI 07-07) and the "Industry Initiative on Management of Materials Issues" (NEI 03-08). These initiatives established frameworks that describe the required activities, standardized approaches, action plans, and assessment criteria used to manage degradation. These initiatives include processes to ensure consistent implementation across the industry. The NRC can perform inspections of activities associated with initiatives to assess the effectiveness of plant performance with respect to the specified guidance. Where the staff has determined that the initiative guidance, scope, goals and schedule for implementation are appropriate, that the implementation of the guidance is effective and consistent, and that new regulatory requirements are not needed, additional staff regulatory actions have not been necessary.

By letter dated November 20, 2009 (ADAMS Accession No. ML093350032), NEI indicated that the nuclear industry's chief nuclear officers voted to approve a proposed "Buried Piping Integrity Initiative." The stated goal of the initiative is to "provide reasonable assurance of structural and leakage integrity of all buried piping with special emphasis on piping that contains radioactive materials." Objectives include proactive assessment and management of the condition of buried piping systems and technology development to improve upon available techniques for inspecting and analyzing underground piping. The initiative implements the EPRI program that is modeled after the NACE Standards. Industry representatives have met among themselves to begin to develop implementation guidelines and other program documents. They have scheduled meetings in January and early February. They will meet with NRC staff at a public meeting in late February to provide more information related to the initiative.

Conclusions

The industry has developed a Buried Piping Integrity initiative that is intended to implement a program that is consistent with guidance provided in NACE Standards. This initiative has the potential to reduce the incidence of leakage from buried piping.

Actions

The staff will work with industry to understand the initiative, including the schedule for action. Also, as discussed previously, the staff will determine whether to perform audits and/or develop a TI to assess licensee implementation of the industry Buried Piping Integrity Initiative.

International Activities

The staff plans to assess international operating experience, standards, and maintenance practices related to buried piping. Staff has proposed to lead this activity through the Nuclear Energy Association's Committee on the Safety of Nuclear Installations (CSNI). This activity falls under a broader international effort to assess issues related to the long-term operation (i.e., more than 60 years) of nuclear plants. The proposed activity on buried piping was approved by CSNI during its semiannual meeting in December 2009. Results from this activity will also be used to assess the need for, and subsequently inform, future regulatory actions.

SUMMARY

Based on the staff's review of operating experience related to buried piping degradation, current regulations and ASME Code requirements have been effective in ensuring that the structural integrity and functionality of buried, safety-related piping are maintained. Current regulations have also been effective in ensuring unintended releases of hazardous material to the environment from leaks in both safety-related and nonsafety-related buried piping remain below regulatory limits. Therefore, the staff has no current plans to recommend regulatory changes to address degradation of buried piping.

The staff will continue to actively participate in ASME Code and NACE standards activities. The revised GALL report, including revisions to the AMPs related to buried piping, is scheduled to be published in December 2010.

The industry has recently developed the Buried Piping Integrity Initiative. The staff plans to meet with the industry to further understand this initiative and monitor industry implementation. The staff will also evaluate the need to revise NRC inspection procedures to assess licensee implementation of this new initiative.

The staff will continue to monitor operating experience and assess the need for any further regulatory actions or communications.

Table 1: Radiological Consequences of Several Recent Buried Piping Degradation Events

Date	Site	Brief Description
August 25, 2009	Oyster Creek, Unit 1	On August 25, 2009 water containing tritium leaked from an underground condensate transfer pipe at Oyster Creek, Unit 1. The aluminum pipe is 6" in diameter (with 0.288" wall thickness) and is not safety related. It had been replaced in 1994 and was examined most recently in April 2009. At the point of the leak, the tritium concentration was approximately 11E6 pCi/l, and the flow rate was approximately 5 gpm. Since the leak was on Oyster Creek property, no member of the public has received any exposure from this leak. No radioactivity from this leak has been detected in any publically accessible area. The water in the ground on-site is not publicly accessible and is not drinking water, so the Environmental Protection Agency (EPA) drinking water standards would not apply. However, EPA would classify drinking water containing 11E6 pCi/l of tritium as exceeding the safe drinking water standards for radionuclides of 20,000 picocuries per liter (pCi/L). Tritium from the leak is being diluted by the ground water and as a result, the remaining tritium concentration is constantly being reduced as it travels underground toward the site boundary. Once the tritium leaves Oyster Creek's property, the projected public doses will be below the Oyster Creek's licensing limits (described in the licensee's Technical Specification and ODCM). As a result, when the tritium leaves Oyster Creek's property, the effluents will conform to NRC's design objectives described in 10 CFR 50, Appendix I. This satisfies the NRC requirement that effluents are ALARA. The NRC is concerned because this release was an abnormal release from an unapproved release point. The licensee is required to include information about this leak in either the licensee's Annual Effluent Report or the Annual Environmental Report. This level of tritium does not exceed any NRC limit (that would be applicable on-site), and after additional dilution and decay it will not exceed any NRC limit at the site boundary or offsite. Although this leak does not exceed any NRC limit, either on-site or offsite, this level of tritium would trigger the licensee to initiate voluntary communications with local and state officials as outlined in the industry Groundwater Initiative, NEI 07-07. During an October 22, 2009, public meeting the licensee indicated future plans to reposition existing risk significant piping to above ground or more accessible locations to enable enhanced monitoring of pipe conditions.

Date	Site	Brief Description
July 9, 2009	Peach Bottom	The results of 6 tritium test wells in the vicinity of the Unit 3 turbine building show the tritium level in 3 wells were > 20,000 pCi/L. The maximum level was 122,748 pCi/L from geo-probe well #4 for a sample drawn on July 8, 2009. The licensee followed the NEI Voluntary Tritium Reporting Initiative in communicating with external stakeholders. There is no information that the tritium is migrating offsite or affecting drinking water sources for site personnel. This level of tritium does not exceed any NRC limit (that would be applicable on-site), and after additional dilution and decay it will not exceed any NRC limit offsite. Although this leak does not exceed any NRC limit, either on-site or offsite, this level of tritium would trigger the licensee to initiate voluntary communications with local and state officials as outlined in NEI 07-07.
June 5, 2009	Dresden	On June 5, 2009 water containing tritium leaked from two underground condensate transfer pipes at Dresden. At (or near) the point of the leaks, the tritium concentration was as high as approximately 3.2E6 pCi/l. Since these leaks were on Dresden property, no member of the public has received any exposure from these leaks. No radioactivity from this leak has been detected in any publically accessible area. The water in the ground on-site is not publicly accessible and is not drinking water, so the EPA drinking water standards would not apply. However, EPA would classify drinking water containing 3.2E6 pCi/l of tritium as exceeding the safe drinking water standards for radionuclides. Tritium from the leak is being diluted by the ground water and as a result, the remaining tritium concentration is constantly being reduced as it travels underground toward the site boundary. Once the tritium leaves Dresden's property, the projected public doses will be below Dresden's licensing limits (described in the licensee's Technical Specification and ODCM). As a result, when the tritium leaves Dresden's property, the effluents will conform to NRC's design objectives described in 10 CFR 50, Appendix I. This satisfies the NRC requirement that effluents are ALARA. The NRC is concerned because this release was an abnormal release from an unapproved release point. The licensee is required to include information about this leak in either the licensee's Annual Effluent Report or the Annual Environmental Report. This level of tritium does not exceed any NRC limit (that would be applicable on-site), and after additional dilution and decay it will not exceed any NRC limit offsite. Although this leak does not exceed any NRC limit, either on-site or offsite, this level of tritium would trigger the licensee to initiate voluntary communications with local and state officials as outlined in NEI 07-07. State of Illinois regulations require

Date	Site	Brief Description
		notification if off-site releases are greater than 200 pCi/l or if concentrations on-site exceed 0.002 curies. The licensee estimated the total amount of tritium on site exceeded 0.002 curies and the state of Illinois was contacted. Dresden experienced similar leaks In 2004 and 2006 and had replaced some underground pipe at that time.
April 15, 2009	Oyster Creek	On April 15, 2009, water containing tritium leaked from two underground condensate transfer pipes at Oyster Creek, Unit 1. Both pipes were carbon steel pipe; one 8" in diameter and the other 10" in diameter. Neither is safety related. Near the point of the leaks, the tritium concentration was approximately 4.5E6 pCi/l. Since the leak was on Oyster Creek property, no member of the public has received any exposure from this leak. No radioactivity from this leak has been detected in any publically accessible area. The water in the ground at the source of the leak on-site is not publicly accessible and is not drinking water, so the EPA drinking water standards would not apply on-site at that location. However, EPA would classify drinking water containing 4.5E6 pCi/l of tritium as exceeding the safe drinking water standards for radionuclides. Tritium from the leak is being diluted by the ground water and as a result, the remaining tritium concentration is constantly being reduced as it travels underground toward the site boundary. Once the tritium leaves Oyster Creek's property, the projected public doses will be below the Oyster Creek's licensing limits (described in the licensee's Technical Specification and ODCM). As a result, when the tritium leaves Oyster Creek's property, the effluents will conform to NRCs design objectives (described in 10 CFR 50, Appendix I). This satisfies the NRC requirement that effluents are ALARA. The NRC is concerned because this release was an abnormal release from an unapproved release point, and as a result, the licensee is required to include information about this leak in either the licensee's Annual Effluent Report or the Annual Environmental Report. This level of tritium does not exceed any NRC limit (that would be applicable on-site), and after additional dilution and decay it will not exceed any NRC limit offsite. Although this leak does not exceed any NRC limit, either on-site or offsite, this level of tritium would trigger the licensee to initiate voluntary communications with local and state officials as outlined in NEI 07-07.

POLICY ISSUE
INFORMATION

March 28, 2010 SECY-10-0034

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: POTENTIAL POLICY, LICENSING, AND KEY TECHNICAL ISSUES FOR SMALL MODULAR NUCLEAR REACTOR DESIGNS

PURPOSE:

To inform the Commission of potential policy, licensing, and key technical issues that may require Commission consideration to support future design and license review applications for small modular reactors (SMRs), and the staff's plans for developing plans for their resolution.[1]

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) staff has been meeting with the Department of Energy (DOE) and, as resources allowed, with individual SMR designers to discuss potential policy, licensing, and key technical issues for SMR designs. As a result of these pre-application activities and earlier work by the NRC staff and Commission, the NRC staff has identified a number of potential policy and licensing issues. The enclosure to this paper provides a summary description of these potential policy issues for Commission information. The discussions are consistent with information provided in previous Commission papers and other related agency documents. The references provided in Attachment 2 to the enclosure include these key Commission documents.

The NRC staff plans to develop proposed resolutions to these potential policy issues and will inform the Commission and other stakeholders of its activities and progress on resolving them. Although approaches to potential resolutions are described, the enclosure does not include proposed resolutions for any of the issues. As information is available and the evaluations

CONTACT: Thomas J. Kenyon, NRO/ARP
(301) 415-1120

[1] A design review application could involve a request for a design approval or design certification under Title 10 of the *Code of Federal Regulations* Part 52 (10 CFR Part 52). A license review application could involve a request for a combined license, manufacturing license, or early site permit under 10 CFR Part 52 or a request for a construction permit and operating license under 10 CFR Part 50.

progress, the NRC staff will prepare future papers that propose potential resolutions or paths to resolution of policy issues to support the Next Generation Nuclear Plant (NGNP) and other SMR review activities. In addition, the staff will inform the Commission in a timely manner of additional issues when they are identified.

BACKGROUND:

As discussed in SECY-08-0019,[2] nuclear reactor designers are developing new nuclear reactor designs and technologies, and have notified the NRC that they may submit design and license applications for some of their SMR designs to the NRC as early as FY 2011. These include (1) a license application for construction and operation of a helium-cooled very-high-temperature reactor in connection with the NGNP project established by the Energy Policy Act of 2005; (2) a design certification (DC) application for the International Reactor Innovative and Secure pressurized-water reactor (PWR) design; (3) a DC application and possible combined license (COL) application for the NuScale Power Reactor PWR design; (4) a DC application for the mPower PWR design; (5) a design approval application for the Super-Safe, Small and Simple sodium-cooled fast reactor (SFR); and (6) prototype COL and manufacturing license applications for the Power Reactor Inherently Safe Module SFR design. Other innovative reactor design and site development activities could lead to the submission of additional design and license review applications for SMRs to the NRC within the next 10 years, but they are not addressed in this paper because of the preliminary status of their development.

The Commission's final policy statement on the regulation of advanced reactors[3] states:

> To provide for more timely and effective regulation of advanced reactors, the Commission encourages the earliest possible interaction of applicants, vendors, other government agencies, and the NRC to provide for early identification of regulatory requirements for advanced reactors and to provide all interested parties, including the public, with a timely, independent assessment of the safety and security characteristics of advanced reactor designs. Such licensing interaction and guidance early in the design process will contribute towards minimizing complexity and adding stability and predictability in the licensing and regulation of advanced reactors.

Furthermore, in the NGNP Licensing Strategy,[4] the Commission stated that in order to implement the NGNP licensing strategy successfully, and meet the congressionally-mandated operation date of 2021, the NRC and DOE needed to implement a pre-application review to identify and resolve policy, regulatory, and key technical issues for the NGNP. Early resolution or identification of a clear path to resolution for issues related to SMRs will enable designers to incorporate appropriate changes during the development of their designs before submitting a design or license review application. Accordingly, the NRC staff has been interacting with DOE on the NGNP and, on a limited basis in accordance with resource availability, with the designers of new SMRs to become familiar with the new designs and technologies, and to provide feedback to DOE and pre-applicants on potential key design, technology, and licensing issues

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[2] SECY-08-0019, "Licensing and Regulatory Research Related to Advanced Nuclear Reactors," dated February 14, 2008. (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091130253 (publicly available), ML073370326 (non-publicly available) and ML073370532 (non-publicly available)). All documents referenced in this paper are available in ADAMS on the NRC's web page (www.nrc.gov) under the accession number provided, except where noted.

[3] Policy Statement on the Regulation of Advanced Reactors: Final Policy Statement, 73 Federal Register 60,612, and 60,616 (October 14, 2008)

[4] "Next Generation Nuclear Plant Licensing Strategy - A Report to Congress," dated August 2008 (ADAMS Accession No. ML082290017)

and on their technology development program plans. These interactions will also provide information to determine NRC infrastructure development and research needs and plans.

DISCUSSION:

The NRC staff has been meeting with DOE and, as resources allowed, with individual SMR designers to discuss potential policy, licensing, and key technical issues for SMR designs. The NRC staff also conducted an SMR workshop in October 2009[5] with SMR designers, DOE, the Nuclear Energy Institute, and other stakeholders to discuss potential policy issues that are common to more than one design. The staff encouraged the participants to work together or with other organizations to generically address issues common to all nuclear designs, SMRs, or specific technology groups (i.e., integral PWRs) in order to focus the issues, propose and obtain consistent resolutions, and effectively use resources. As a result of these pre-application activities and earlier work by the NRC staff and Commission, the NRC staff has identified a number of potential policy and licensing issues based on the preliminary design information provided by pre-applicants and discussions with the designers and DOE regarding their proposed approaches to addressing key issues. The enclosure describes those potential policy issues that the staff has identified. In general, these issues result from the key differences between the new designs and current-generation pressurized-water reactors (such as size, moderator, coolant, fuel design, and projected operational parameters), but they also result from industry-proposed review approaches and industry-proposed modifications to current policies and practices. This paper addresses only those potential policy and licensing issues for which resolutions may require Commission consideration. It does not address key technical issues related to these designs unless their importance to the design and the potential impact of policy issue resolutions require such discussion. The description and references provided in the enclosure are not intended to be all inclusive. In addition, although approaches to potential resolutions are described, the enclosure does not include proposed resolutions for any of the issues.

The NRC staff plans to develop proposed resolutions to these issues by continuing to obtain information from DOE, potential design and license applicants, and other sources (both domestic and international); identifying and developing proposals for the resolution of policy issues; and where appropriate, preparing papers proposing resolutions of these issues with recommendations for consideration and approval by the Commission. Although the staff discusses a number of potential policy issues concerning SMRs in the enclosure, it has identified some key issues that it considers most important to resolve by FY 2011 or FY 2012 in order to support the design development of the NGNP and integral PWRs. The following is a brief description of these key issues. They are discussed in greater detail in the enclosure along with the other potential policy issues that may need to be addressed as the NRC staff conducts its SMR reviews.

Implementation of the Defense-In-Depth Philosophy for Advanced Reactors

The Commission has had a long-standing policy of ensuring that defense-in-depth (DID) is incorporated into the design and operation of nuclear power plants. The requirements in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," incorporate DID measures specific to light-water reactors (LWRs). Although integral PWRs employ the more traditional DID approach of LWRs in their designs, non-LWR SMR designers propose to use

[5] "Summary of Workshop on Small- and Medium-Sized Nuclear Reactors (SMRs)," U.S. Nuclear Regulatory Commission, October 22, 2009. (ADAMS Accession No. ML092940138)

different approaches to establish DID barriers for their designs. This can be seen in their approaches to address technical issues such as redundancy of key safety-related components and containment functional capability. The DID measures have been determined on a case-by-case basis for non-LWRs licensed in the past. Preventive or mitigative compensatory measures may need to be incorporated into the design or operation of certain SMRs to account for uncertainties in design or operational capability of the facility. In FY 2010 and FY 2011, the NRC staff will review pre-application submittals concerning DID that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, consider approaches to DID proposed by the domestic and international community, and determine whether preventive or mitigative compensatory measures may be needed for SMR designs to account for uncertainties in design or operational capability of the facility. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning DID in FY 2011 to support development of the NGNP or other SMR designs.

Appropriate Source Term, Dose Calculations, and Siting for SMRs

Accident source terms are used for the assessment of the effectiveness of the containment and plant mitigation features, site suitability, and emergency planning. Other radiological source terms are used to show compliance with regulations on dose to workers and the public. Design and license applicants and the NRC will need to establish appropriate bounding source terms for high-temperature gas-cooled reactors and other SMRs. There may also be source-term issues associated with the multi-module aspect of SMRs where modules share structures, systems, and components (SSCs). For example, the Commission may have to determine when it would be appropriate to base the bounding source term on an accident in a single module and when possible sharing of SSCs require the evaluation of core damage in and potential releases from more than one module. In FY 2010 and FY 2011, the NRC staff will review pre-application submittals concerning source-term issues that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulations or propose new regulatory guidance concerning the source term and site suitability for an SMR in FY 2011 to support development of the NGNP and other SMR designs.

Appropriate Requirements for Operator Staffing for Small or Multi-Module Facilities

Some SMR designs may use multiple modules at one site, but current regulations do not address the possibility of more than two reactors being controlled from one control room. In addition, SMR designers have indicated that they are considering whether their designs can operate with a staffing complement that is less than that currently required by the Commission's regulations. Other potential SMR policy issues include the possible need for requirements on control room staffing during refueling operations, reactor staff who interact with an interconnected manufacturing plant, supervisory staff, shift work, and training. In FY 2010 and FY 2011, the NRC staff will review pre-application submittals concerning operator staffing and associated control room design that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, discuss the proposed resolutions with human factors and instrument and controls experts, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulatory guidance or staff positions or propose new guidance concerning the operator staffing for an SMR in FY 2012 to support development of the NGNP and other SMR designs.

Security and Safeguards Requirements for SMRs

Because many SMRs are still in early developmental stages and the designs are not yet fixed, SMR designers have a unique opportunity to determine the appropriate design basis threat; develop emergency preparedness; and integrate physical security protection, cyber security protection, and material control and accounting (MC&A) measures with the design and operational requirements during the design process and during the development of the a license applicant's physical security and MC&A programs and systems. SMR designers are expected to integrate security into the design and will need to conduct a security assessment to evaluate the level of protection provided, including safeguards aspects of SMR-related fuel cycle and transportation activities. The DOE, SMR designers, and potential operators have raised issues regarding the appropriate number of security staff and size of the protected area. In FY 2010 and FY 2011, the NRC staff will review pre-application submittals concerning safeguards that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, discuss the proposed resolutions with safeguards experts, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning safeguards for an SMR in FY 2011 to support development of the NGNP and other SMR designs.

The staff is developing detailed resolution plans for each issue discussed in this paper, taking into account factors such as whether resolution of the issue is critical to the development of the NGNP or integral LWR designs; the number of affected technology groups and design centers; the potential effect on design decisions; the potential need for legislation, rulemaking, or policy changes; the potential need for confirmatory research; the participation and cooperation of applicants, other Government agencies, professional societies, and other stakeholders; the potential effect on the schedule for prototype plants or commercial deployment; and the dependencies on other policy or technical issues (e.g., development of source-term models). The staff will refine and implement the resolution plans for each issue as it receives additional information from DOE, pre-applicants or applicants, or other sources in FY 2010 and FY 2011, and as the staff assesses possible solutions to the technical and policy issues. The staff will address technical issues using established processes, including public participation, for issuing regulatory guidance, and will provide future papers to the Commission describing the proposed resolutions and the NRC staff positions and recommendations regarding each of the major policy issues. The staff will provide information to the Commission and other stakeholders regarding its activities and progress on resolving the policy and key technical issues using established mechanisms such as public meetings, postings on the NRC web page, and routine reporting vehicles such as the quarterly updates on the status of new reactor review activities.

RESOURCES:

The resources allocated to conduct the activities described in this paper (including those for supporting offices) are included in budgeted activities listed below related to the reviews of SMRs. There is $14.2M, including 29.4 full time equivalents (FTEs) budgeted in FY 2010. There is $18.8M, including 49 FTE, included in the FY 2011 Presidents Budget. The resources for FY 2012 and beyond will be requested using the planning, budgeting, and performance

management process as the staff better understands the complexity of these issues and their effect on the SMR designs.

	FY 2010			FY 2011		
	Contract $	Total FTE	Amount	Contract $	Total FTE	Amount
Total	$9,756	29.4	$14,195	$11,430	49.0	$18,819
NRO	3,166	12.9	5,114	5,994	26.2	9,945
NSIR	0	0.5	75	0	1.8	271
RES	6,590	16.0	9,006	5,436	21.0	8,603

CONCLUSIONS:

The NRC staff will continue its pre-application activities on the NGNP and its interactions with the designers of other SMRs to further identify and resolve policy, licensing, and key technical issues. The staff is developing detailed resolution plans for each issue. As the plans are implemented, the staff will prepare papers that propose resolutions or paths to resolution of policy issues to support the NGNP and other SMR review activities. In addition, the staff will inform the Commission in a timely manner of additional issues when they are identified.

COORDINATION:

This paper has been coordinated with the Office of the General Counsel, which has no legal objection, and with the Office of the Chief Financial Officer.

/RA by Bruce S. Mallett for/

R. W. Borchardt
Executive Director
for Operations

Enclosure:
Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs

POTENTIAL POLICY, LICENSING, AND KEY TECHNICAL ISSUES FOR SMALL MODULAR NUCLEAR REACTOR DESIGNS

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Enclosure

POTENTIAL POLICY, LICENSING, AND KEY TECHNICAL ISSUES FOR SMALL MODULAR NUCLEAR REACTOR DESIGNS

1.0 Introduction

The U.S. Nuclear Regulatory Commission (NRC) staff has been conducting pre-application interactions with the U.S. Department of Energy (DOE) on the Next Generation Nuclear Plant (NGNP) and, on a limited basis in accordance with resource availability, with the designers of other new small modular nuclear reactors (SMRs). The NRC staff has identified a number of potential policy and licensing issues that may require resolution during the review of design and license applications[1] for some of these designs. The reactor designs included in this review include the following: [2]

- DOE's NGNP, a helium-cooled very-high-temperature reactor (VHTR)
- Pebble Bed Modular Reactor (Pty) Limited's Pebble Bed Modular Reactor (PBMR), a 400 megawatt-thermal (MWt) (165 megawatt-electric (MWe)) pebble bed gas-cooled reactor design
- General Atomics' Gas-Turbine Modular Helium Reactor (GT-MHR), a 600 MWt (285 MWe) prismatic gas-cooled reactor design
- AREVA NP, Inc.'s AREVA's New Technology Advanced Reactor Energy System (ANTARES), a 600 MWt (285 MWe) prismatic gas-cooled reactor design
- Westinghouse Electric Company's International Reactor Innovative and Secure (IRIS), a 1000 MWt (335 MWe) pressurized light-water reactor (PWR) design with an integral nuclear steam supply system (NSSS)
- NuScale Power, Inc.'s NuScale Power Reactor (NuScale), a 160 MWt (45 MWe) natural-circulation PWR design with an integral NSSS
- Babcock & Wilcox Company's mPower reactor design, a 400 MWt (125 MWe) PWR design with an integral NSSS
- Toshiba Corporation's Super-Safe, Small and Simple (4S) 30 MWt (10 MWe) sodium-cooled fast reactor (SFR) design
- GE Hitachi Nuclear Energy's Power Reactor Inherently Safe Module (PRISM) 471 MWt (155 MWe) SFR design

[1] A design review application could involve a request for a design approval or design certification under Title 10 of the *Code of Federal Regulations* Part 52 (10 CFR Part 52). A license review application could involve a request for a combined license, manufacturing license, or early site permit under 10 CFR Part 52 or a request for a construction permit and operating license under 10 CFR Part 50.

[2] The power levels presented represent nominal values of the reference designs to provide information on the size of the designs. The actual design values may change as the design is finalized. Additional descriptions of the designs and a discussion of the interrelationship between the NGNP and other gas-cooled reactor designs are provided in Attachment 1 to this enclosure.

Other reactor design development activities could result in evaluation of other SMR designs sometime in the next 10 years, but they are not addressed in this paper because of the preliminary status of their design development.

As a result of its pre-application activities and earlier work by the NRC staff and Commission, the NRC staff has identified a number of potential policy, licensing, and key technical issues based on review of the preliminary design information provided by the pre-applicants and discussions with the designers and DOE regarding their proposed approaches to addressing key issues. In developing this list of issues, the NRC staff considered the following:

- policy issues previously identified to the Commission,
- the unique aspects of these designs,
- the applicability of current regulatory requirements and guidance to these designs,
- its previous and current reviews of light-water reactor (LWR) and non-LWR designs,
- international experience with licensing and operation of advanced reactor designs,
- operating experience of commercial, test, and research reactors, and
- the results of available probabilistic risk assessments (PRAs).

The NRC staff has met with individual SMR designers to discuss potential policy, licensing, and key technical issues for their specific designs. The NRC staff also conducted an SMR workshop on October 8-9, 2009, to discuss potential policy issues that are common to more than one design with SMR designers, DOE, the Nuclear Energy Institute (NEI), and other stakeholders. The staff encouraged the participants to work together or with other industry organizations to generically address issues common to all nuclear designs, SMRs, or specific technology groups (i.e., integral PWRs) in order to focus the issues, propose and obtain consistent resolutions, and effectively use resources. Early resolution or identification of a clear path to resolution for these issues will enable SMR designers to incorporate appropriate changes during the development of their designs before submitting a design or license application.

In general, these issues result from the key differences between the new designs and current-generation LWRs (such as size, moderator, coolant, fuel design, and projected operational parameters), but they also result from industry-proposed review approaches and industry-proposed modifications to current policies and practices. As indicated earlier, some of these issues are common to all nuclear reactor designs, and may be resolved in connection with consideration of these issues for all reactors. The following sections describe those issues that the NRC staff considers to be potential policy and licensing issues that the agency will likely have to address while determining the acceptability of these unique designs during design and license reviews, should an application be submitted. This paper does not address key technical issues related to these designs unless their importance to the design and the potential impact of policy issue resolutions require such discussion. The references provided in Attachment 2 of this enclosure are key Commission papers and other documents that address these issues.[1] The description and references provided in the attachment are not intended to be all inclusive, and will be further discussed in future papers, as necessary.

[1] Attachment 2 provides the full title, date issued, and ADAMS accession number for all references.

The staff is developing resolution plans for each issue discussed in this paper taking into account factors such as whether resolution of the issue is critical to the development of the NGNP or integral LWR designs; the number of affected technology groups and design centers; the potential effect on design decisions; the potential need for legislation, rulemaking, or policy changes; the potential need for confirmatory research; the participation and cooperation of applicants, other Government agencies, professional societies, and other stakeholders; the potential effect on the schedule for prototype plants or commercial deployment; and the dependencies on other policy or technical issues (e.g., development of source term models). The staff will refine and implement the resolution plans for each issue as it receives additional information from DOE, pre-applicants or applicants, or other sources in FY 2010 and FY 2011, and as the NRC staff assesses possible solutions to the technical and policy issues.

The NRC staff is providing an initial characterization of the issues in terms of scope (generic or specific technology group/design center), importance, and likely timing for subsequent Commission papers (FY 2011, FY 2012, or FY 2013 or beyond). The specific resolution plans developed for the issues may change as additional information is collected and assessed. In addition, activities undertaken by the industry or other stakeholders may inform the NRC staff in developing resolution plans and could revise the above initial characterization. For example, the American Nuclear Society has created a special committee to prepare possible positions on various regulatory issues related to small- and medium-sized reactors. This activity, as well as those of other groups and designers, are in preliminary stages and will likely affect both the scope and timing of the resolution plans being developed by the NRC staff. The staff will provide information to the Commission and other stakeholders regarding its activities and progress on resolving the policy and key technical issues using established mechanisms such as public meetings, postings on the NRC web page, and routine reporting vehicles such as the quarterly updates on the status of new reactor review activities.

2.0 Licensing Process Issues for Small Modular Nuclear Reactors

2.1 License for Prototype Reactors

Scope: Design specific
Importance: High
Issue Paper: FY 2013 or beyond

If the progress of an SMR research and development (R&D) program does not fully support an NRC decision on a license application for the proposed commercial version of the design, the design or operation of the first unit may need to include preventive or mitigative compensatory measures to account for uncertainties in the design or operational capability (see 10 CFR 50.43(e)). In addition, the NRC may require special confirmatory tests and measurements in the license in order to confirm that the facility operates in accordance with the designer's analyses. License conditions could be imposed and/or features added to the plant to increase safety margin until such time as the operation of the prototype unit or other testing programs confirm certain aspects of the design and equipment performance. These license conditions could, for example, limit the plant to less than full power, place restrictions on operational temperature, or require more extensive startup or operational testing. Another alternative could be to use initial plant startup as a means to test and confirm plant safety features in lieu of conducting R&D before plant licensing. If such an alternative is chosen, the

scope and nature of the startup or operational test program would need to be agreed upon, but this alternative could involve an incremental licensing approach during startup operations, with power and temperature uprates allowed when confirmatory measurements of core temperature and plant parameters confirm design expectations and predictions. License applicants and the NRC staff have not relied on the construction and operation of a licensed prototype reactor to confirm design assumptions or to even supplement pre-licensing R&D since the early period of the evolution of commercial nuclear power plants. The use of these provisions in NRC regulations may involve policy issues for Commission consideration. The NRC staff also discussed this issue in SECY-02-0180.

This issue was raised as a potential issue for the NGNP in the August 2008 Licensing Strategy, but the staff believes that it could also be applicable to other new, first-of-a-kind designs. The staff believes that resolution for this issue need not occur until after a license application is submitted because the extent of necessary preventive or mitigative compensatory measures and confirmatory testing needs for a prototype will not be known until after the staff has reviewed the applicant's demonstration test program for the design and the proposed operational test program that supports the license. Once a license application is received, the NRC staff will review the prototype design, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and determine whether compensatory measures are needed for the design to account for uncertainties in design or operational capability of the facility. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning the license for the prototype in a timeframe consistent with the licensing schedule.

2.2 License Structure for Multi-Module Facilities

Scope: Generic
Importance: Medium
Issue Paper: FY 2013 or beyond

Issues with the written structure of a design certification or license for multi-module facilities may arise during the review of the application for a modular reactor design, particularly when one module can begin operation while other modules are being built and installed. For example, the NuScale Power reactor design, which could be a multi-module facility, raises issues pertaining to the effective duration of a combined license (COL) issued for such modular reactor designs. Section 52.103(g) of 10 CFR states: "[i]f the combined license is for a modular design, each reactor module may require a separate finding [that the acceptance criteria of the COL are met] as construction proceeds." In the case of NuScale where the designer plans to submit a design certification application for a 12-module facility, a single module may be put into operation, but the other modules may not be put into operation for a significant amount of time, depending on factors such as resource limitations, the need for power, or upgrades to transmission lines. In addition, it is possible that an applicant may submit an application for design certification of a facility that can employ a single reactor or can consist of multiple reactor modules. The license of other SMRs, such as the mPower design, may also be affected, depending on how the applicant submits the license application. Although 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," addresses some aspects of modular facilities, the use

of these provisions may involve policy issues or identify possible regulatory changes requiring Commission consideration and approval.

Although resolution of these issues before submittal of a design certification or license application may be more important to an SMR license applicant trying to support its business case at the design certification stage, the staff believes that resolution of these issues need not occur until after a licensing application is submitted because it concerns activities that will need to be addressed during an operating license review. Once a license application for a multi-module facility is received, the NRC staff will review the application, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, and discuss design-specific proposals to address this matter. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning the license for the multi-module facility in a timeframe consistent with the licensing schedule.

2.3 Manufacturing License Requirements for Future Reactors

Scope: Generic
Importance: Low
Issue Paper: FY 2013 or beyond

The NRC staff has identified a potential policy issue regarding whether a manufacturing license would be allowed or possibly required in addition to a design certification. There are likely jurisdictional issues with respect to requiring and issuing a manufacturing license if the manufacturing is taking place in a foreign country. For example, the PBMR could be fabricated in South Africa and the Toshiba 4S could be manufactured in Japan. B&W plans to fabricate its mPower modules offsite in its U.S. and Canadian facilities, using its integrated manufacturing infrastructure. NuScale Power currently plans to fabricate the modules offsite within the United States, and ship the reactor vessel and steel containment in pieces to the site. The NRC staff may need to consider conditions on an import license with respect to access by NRC inspectors to verify compliance of reactors manufactured outside of the United States.

Also, the regulations for a manufacturing license granted in accordance with 10 CFR Part 52 are structured for a complete facility, including the NSSS and balance-of-plant (BOP). This regulatory structure reflects the only experience the NRC has had with reviewing and issuing a manufacturing license (i.e., Offshore Power Systems' ML-1 for the Floating Nuclear Power Plant, issued in 1982). Issuing a manufacturing license authorizing the manufacture and transport of only major portions of the plant (e.g., the NSSS) and combining these with structures and systems built at specific sites may involve potential policy issues that would require Commission consideration.

Although the PBMR, Toshiba 4S, PRISM, mPower, and NuScale reactors are all candidates for a manufacturing license because of size, manufacturing plans and location, and transportation considerations, only GE-Hitachi currently proposes to submit a manufacturing license for its PRISM SFR design. The staff is currently directing its focus on issues concerning the NGNP and integral PWRs. Therefore, the NRC staff has assigned a low priority to resolution of this issue.

3.0 Issues Concerning Design Requirements for Small Modular Nuclear Reactors

3.1 Implementation of the Defense-In-Depth Philosophy for Advanced Reactors

Scope: Generic (although more germane to non-LWRs)
Importance: High
Issue Paper: FY 2011

The Commission has had a long-standing policy of ensuring that defense-in-depth (DID) is incorporated into the design and operation of nuclear power plants. The requirements in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," incorporate DID measures specific to LWRs (e.g., a pressure-retaining, low-leakage containment). Although integral SMRs employ the more traditional DID approach of LWRs in their designs, non-LWR SMR designers propose to use different approaches to establish DID barriers for their designs. This can be seen in their approaches to address technical issues such as redundancy of key safety-related components and containment functional capability. For non-LWRs licensed in the past (e.g., Fort St. Vrain), DID measures have been determined on a case-by-case basis. Preventive or mitigative compensatory measures may need to be incorporated into the design or operation of certain SMRs to account for uncertainties in design or operational capability of the facility. Therefore, the NRC staff will need to determine appropriate DID measures and develop appropriate requirements and guidance to support design and license reviews of integral PWRs and non-LWR designs.

In SECY-09-0056, the NRC staff stated that it plans to integrate its position on DID with its positions on other policy and key technical issues for future reactor designs during its reviews. The staff plans to continue development of a position on DID along with development of other related Commission policy and technical positions, but it will defer activities to finalize a DID policy statement until it has gained additional experience and related insights from the NGNP or other non-LWR reviews.

The NRC staff believes that resolution of this issue is required to support the design development of the NGNP and potentially other SMR designs. Therefore, it has been assigned a high importance that should be addressed before submittal of the NGNP COL application. In FY 2010 and FY 2011, the NRC staff will review pre-application white papers and topical reports concerning DID that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, consider approaches to DID proposed by the domestic and international community, and determine whether preventive or mitigative compensatory measures may be needed for SMR designs to account for uncertainties in design or operational capability of the facility. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning DID in FY 2011 to support development of the NGNP or other SMR designs.

3.2 Use of Probabilistic Risk Assessment in the Licensing Process for SMRs

Scope: Generic (although more germane to non-LWRs)
Importance: High
Issue Paper: FY 2013 or beyond

In the August 2008 NGNP licensing strategy, the Commission concluded that the best option for licensing the NGNP prototype would be to use a risk-informed and performance-based technical approach that employs the use of deterministic judgment and analysis, complemented by NGNP-specific PRA information. This licensing approach would, where possible, adapt the existing LWR technical requirements to address the acceptability of the NGNP design and establish requirements unique to the NGNP for those technical areas that existing LWR requirements and guidance do not address. The Commission concluded that once NGNP technology is successfully demonstrated through operation and testing of the NGNP prototype, and a quality PRA that includes data from operation of the prototype becomes available, greater emphasis on a design-specific PRA to establish the licensing basis and requirements will be a more viable option for licensing a commercial version of the NGNP reactor.

Design development and possible review approaches have been discussed with the NRC and proposed in other forums (i.e., draft consensus standards and international technical reports) that would place greater emphasis on the use of risk insights to identify licensing basis events and establish the safety classification of systems, structures, and components (SSCs) for reactor designs. This approach is consistent with a licensing approach described in SECY-03-0047 and approved by the Commission in its staff requirements memorandum (SRM) of June 26, 2003. However, in SECY-09-0056, the NRC staff discussed its plans to follow an approach consistent with the NGNP Licensing Strategy for licensing the prototype reactor while also testing and refining requirements and guidance for increased use of risk insights in the licensing process. Should an applicant submit a design for a facility license that uses an approach applying increased use of risk insights to establish the licensing basis before this effort is undertaken and evaluated, the use of this approach may involve policy issues requiring Commission consideration.

In addition, a number of issues related to the application of current risk-informed programs have been raised because of the lower risk estimates for the large LWRs currently under review. The two most common risk metrics used in current risk-informed applications are based on a core damage frequency (CDF) of 10^{-4}/year and a large, early release frequency (LERF) of 10^{-5}/year as surrogates for the Commission's quantitative health objectives. Risk estimates for new reactors are several orders of magnitude (1 to 3 for CDF; and 1 to 4 for radionuclide release frequency) lower than those for current designs when including internally initiated events and those externally initiated events that have been quantified. The lower risk values create challenges regarding how to apply acceptance guidelines for changes to the licensing basis and thresholds in the Reactor Oversight Process (ROP). The NRC staff provided a white paper to the Commission on February 12, 2009, that identifies the issues posed by the lower risk estimates for large LWRs in risk-informed applications and potential options for implementation. On March 27, 2009, NEI submitted its own white paper recommending no change to the current risk metrics. The NRC staff held a meeting to discuss these issues with stakeholders on September 29, 2009, and is drafting a Commission paper to discuss the issue and present policy options to the Commission. These issues are expected to be applicable to integral PWRs

as well. However, these risk metrics are not applicable to non-LWR SMRs, so the NRC will need to determine what risk metrics should be used for changes to the licensing basis and thresholds in the ROP for those designs.

Because the NRC has chosen to use a risk-informed and performance-based technical approach that employs the use of deterministic judgment and analysis, complemented by design-specific PRA information to review the first NGNP, resolution of this issue is not required to conduct the COL review described in the NGNP Licensing Strategy. In addition, the staff plans to employ a similar approach to review design and license applications for integral PWR designs. Therefore, the staff believes that resolution of this issue need not occur until after design or licensing applications are submitted that propose a review approach be used by the NRC staff that places greater emphasis on a design-specific PRA to establish the licensing basis and requirements.

3.3 Appropriate Source Term, Dose Calculations, and Siting for SMRs

Scope: Generic
Importance: High
Issue Paper: FY 2011

Accident source terms are used for the assessment of the effectiveness of the containment and plant mitigation features, site suitability, and emergency planning. Other radiological source terms are used to show compliance with regulations on dose to workers and the public. The Commission has previously deliberated on the use of design-specific and event-specific source terms, provided there was sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment was used to bound uncertainties. The source terms for the integral PWRs may be based partly on source term information from current generation LWRs and insights gained from extensive state-of-the-art fission product experiments conducted to understand accident phenomena including fission product transport and release. The staff will assess what will be necessary to establish the basis for a scenario-specific approach and how uncertainties should be taken into account. In addition, design and license applicants and the NRC will need to establish appropriate bounding source terms for high-temperature gas-cooled reactors (HTGRs) and SFRs. This is discussed in more detail in Section 3.4 of this paper.

There may be regulatory issues that the Commission may have to consider regarding whether the site boundary dose acceptance criteria and associated dose calculations for use in evaluation of site suitability and emergency planning for SMR designs should be updated or amended, or whether new requirements should be established for SMRs. Current regulatory practice employs the siting dose criteria in 10 CFR 50.34 and 10 CFR Part 52 in conjunction with deterministic design basis accident analyses as the key input parameters for analyzing the effectiveness of the containment, determining site suitability, and preparing site emergency plans.

As discussed in the footnotes in 10 CFR 52.79(a), the current regulations on siting are based on deterministic evaluation of a large fission product release from a substantially melted core to an intact containment, with design leakage to the environment and calculation of cumulative dose to a reference person at two different locations offsite. These accident assumptions may not be

applicable for some SMR designs, which may call into question the applicability of the dose criteria as well.

In addition to the appropriate source terms for the SMR designs, the evaluation of site suitability may include consideration of the population density; use of the site environs, including proximity to man-made hazards; and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology for the SMR designs. Therefore, there may be regulatory issues that the Commission may have to consider regarding whether the seismic and geologic siting criteria and earthquake engineering criteria should be updated or amended, or whether new requirements should be established for SMRs to incorporate advancements of earth science and earthquake engineering for use in evaluation of the site suitability for some SMR designs.

There may also be source-term issues associated with the multi-module aspect of SMRs where modules share SSCs. For example, the Commission may have to determine when it would be appropriate to base the bounding source term on an accident in a single module and when could possible sharing of SSCs require the evaluation of core damage in and potential releases from more than one module. Issues related to source term and risk evaluations for multi-module facilities may relate to policy and therefore, require Commission consideration.

The NRC staff believes that resolution of this issue is required to support the design development of the NGNP. Interrelated issues could also affect the design of integral PWRs. Therefore, it has been assigned a high importance that should be addressed before submittal of design or license applications of these technology groups. In FY 2010 and FY 2011, the NRC staff will review pre-application white papers and topical reports concerning source-term issues that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulations or regulatory guidance or propose new guidance concerning the source term for an SMR in FY 2011 to support development of the NGNP, integral PWRs, or other SMR designs.

3.4 Key Component and System Design Issues for SMRs

This subsection provides examples of how resolutions to policy issues could impact key component and system design technical issues. When the time is right, the staff will present these to the Commission for a decision. At the time the issues are presented and the Commission has determined the appropriate resolution or resolution paths for the policy issues described in this paper, the NRC staff will address how these resolutions should be applied to address technical issues, using established processes, including public participation, for issuing regulatory guidance. In addition, the NRC staff expects that SMR applicants will provide information to the NRC staff showing how policy issue resolutions have been applied to addressing key technical issues when they submit design and license review applications, and the NRC staff will evaluate the acceptability of the implementation of the policy decisions in the designs. The NRC staff does not anticipate that these technical issues will require Commission consideration provided that the resolutions to SMR policy issues are appropriately implemented. However, because of their importance to the design and the potential impact of resolutions of

policy issues, they are provided to provide the Commission context regarding the affect of applicable policy issue resolutions.

The technical issues affected by the resolutions of the policy issues in this paper include:

- ***Core Composition and Source Term Issues for SMRs***

  Scope: Technology/design-specific
  Importance: High

  As discussed in Section 3.3 of this paper, source terms are used for the assessment of the effectiveness of the containment and plant mitigation features, site suitability, and emergency planning. The source terms for the integral PWRs may be based partly on source-term information from current generation LWRs and insights gained from extensive state-of-the-art fission product experiments conducted to understand accident phenomena including fission product transport and release. In addition, license applicants and the NRC will need to establish appropriate bounding source terms for HTGRs and SFRs and the conditions under which their use can be justified in licensing.

  In SECY-93-0092, the NRC staff proposed that source terms for HTGRs and SFRs should be based upon a bounding mechanistic analysis that meets certain performance and modeling criteria supported by research and test data. The conditions under which the use of design-specific and event-specific mechanistic source terms can be justified and used in licensing non-LWRs will have to be supported by experimental data to confirm the parameters of the source term. In its SRM dated July 30, 1993, the Commission approved the staff's recommendation. The NRC staff will ensure that uncertainties are accounted for in the designs. Because of the implications of using design-specific and event-specific mechanistic source terms in licensing, the technical basis for and the uses of such source terms in licensing are critical to the resolution of this technical issue.

  In addition, differences in the core composition of non-LWRs could result in potential policy issues concerning fuel cycle and transportation impacts, including environmental impacts of the production, transportation, and storage of reactor fuel and radioactive waste for non-LWRs. In SECY-02-0180, the NRC staff recommended that the environmental effects of the production, transportation, and storage of reactor fuel and radioactive waste be reviewed on an application-by-application basis for non-LWR license applicants. The Commission approved the staff's recommendation in its SRM dated March 31, 2003.

- ***Accident Selection for SMRs***

  Scope: Generic (although more germane to non-LWRs)
  Importance: High

  For SMRs, the NRC staff will need to consider a different or revised set of accidents than those considered for current LWRs to provide a basis for selecting a mechanistic siting source term and for judging the adequacy of features such as containment functional design and offsite emergency planning. The NRC staff will need to consider accident scenarios during power ascension, full power operation, power decrease, and low power operations.

In the August 2008 NGNP Licensing Strategy, the Commission stated that licensing-basis event categories (i.e., abnormal occurrences, design-basis accidents, and beyond-design-basis accidents) would be established based on the expected probability of event occurrence. However, selection of licensing basis events within each category would be performed using deterministic engineering judgment complemented by insights from the NGNP PRA. In general, the NRC staff expects to apply this approach to all SMRs.

Although identification of many accident scenarios will likely be straightforward, the application of certain scenarios may require Commission consideration. For example, designers of HTGRs have previously proposed that the failure of the vessel or piping connecting the reactor vessel and steam generator vessel need not be considered as a design basis event. In addition, although the Commission has previously stated that certain events should be addressed for non-LWR designs, subsequent research and evaluations may challenge the need to analyze these low probability events.

- ***Redundancy of the Passive Residual Heat Removal System***

Scope: Generic
Importance: High

In SECY-93-0092, the NRC staff identified an issue regarding whether advanced reactor designs that rely on a single, completely passive, safety-related residual heat removal (RHR) system would be acceptable. The staff stated that the unique features of the PRISM and Modular High-Temperature Gas-Cooled Reactor (MHTGR) designs lead the NRC staff to believe that reliance on such an RHR system may be acceptable, depending on how the designer addresses this issue. In performing its detailed design evaluation, the NRC staff committed to ensure that NRC regulatory treatment of non-safety-related backup RHR systems is consistent with Commission decisions on passive LWR design requirements. In its SRM dated July 30, 1993, the Commission approved the staff's approach. The NRC staff will ensure that treatment of proposed non-safety-related backup systems is adequately addressed in SMR designs.

- ***Classification of Structures, Systems, and Components***

Scope: Generic (although more germane to non-LWRs)
Importance: High

During its reviews of recent LWR design and license applications, the NRC staff has used deterministic judgment, complemented by insights from the design-specific PRA, to review SSCs relied on to prevent or mitigate safety-significant licensing-basis events. In conducting its review, the staff verified that safety margins were adequate to ensure the integrity and performance of safety-significant SSCs using a conservative analysis or a best-estimate analysis with consideration of uncertainties. The NRC staff expects to apply this approach to most of the SMR design reviews. If necessary, special treatment requirements would be established to ensure the required performance capability and reliability of the safety-significant SSCs, using deterministic engineering judgment, complemented by insights and information from the design-specific PRA.

The NRC staff stated that it planned to use this approach to classify the SSCs for the NGNP in the August 2008 NGNP Licensing Strategy. However, as discussed in Section 3.2 of this paper, alternative approaches are being considered that put more emphasis on the use of risk insights that are complemented by deterministic evaluations and engineering judgment. DOE or an SMR designer may propose such an approach to justify modification of the design, installation, and maintenance requirements of the identified safety-related SSCs. Once that policy issue is resolved, the NRC staff will ensure that it is adequately implemented when conducting its design or license reviews.

- ***Containment Functional Capability for SMRs***

Scope: Generic (although more germane to non-LWRs)
Importance: High

Fission product retention during an accident involving an HTGR will be highly dependent upon the ability of its coated fuel particles to maintain their integrity and retain fission products during normal operation and accident conditions. Previous gas-cooled reactor designs have relied on similar coated fuel particle technology and have demonstrated the feasibility of using fuel as the primary barrier to fission product release. SFR designers rely on their fuel characteristics and cladding, the reactor vessel, and a containment system that is expected to be exposed to low pressures during an accident to provide multiple barriers to retain fission products. The IRIS and mPower LWR designs employ more conventional LWR barrier designs, relying on their fuel cladding, the reactor coolant pressure boundary, and containment design to retain fission products, and are not expected to raise policy issues in this area. However, the NuScale LWR design employs a non-traditional, small containment for each module that operates in a large pool of water. This unique design could raise construction and operational issues that must be adequately addressed by the designer.

In SECY-03-0047, the NRC staff recommended that the Commission approve the use of functional performance requirements to establish the acceptability of a containment or confinement structure (i.e., consideration of a non-pressure-retaining building provided certain performance requirements can be met). In developing the requirements for SMRs, the need for and type of containment barrier will have to be established. This will involve taking into consideration factors such as fuel quality and performance, plant transient behavior, security, aircraft impact assessments, and DID.

In an SRM to SECY-03-0047, the Commission disapproved the staff's recommendation related to the requirement for a pressure-retaining containment building, but directed the staff to pursue the development of functional performance standards and then submit options and recommendations to the Commission on this issue. The variety of designs currently being proposed may result in this issue being brought before the Commission for resolution on specific designs or groups of designs.

4.0 Operational Issues for Small Modular Nuclear Reactors

4.1 Appropriate Requirements for Operator Staffing for Small or Multi-Module Facilities

Scope: Generic
Importance: High
Issue Paper: FY 2012

Some SMR designs may use multiple modules at one site, but current regulations do not address the possibility of more than two reactors being controlled from one control room. In SECY-93-0092 and SECY-02-0180, the NRC staff discussed whether advanced reactor designs should be allowed to control more than two reactors from one control room and operate with a staffing complement that is less than that currently required by the Commission's regulations. The NRC staff stated that it believed that operator staffing may be design dependent and intended to review the justification for a smaller crew size for the advanced reactor designs by evaluating the function and task analyses for normal operation and accident management. In SECY-93-0092, the staff identified several factors that could be used in assessing the staffing levels for SMRs, including the following:

- Whether smaller operating crews could respond effectively to a worst-case array of power maneuvers, refueling and maintenance activities, and accident conditions.

- Whether an accident at a single unit could be mitigated with the proposed number of licensed operators, less one, while all other units could be taken to a cold-shutdown condition from a variety of potential operating conditions, including a fire in one unit.

- Whether the units could be safely shut down with eventual progression to a safe shutdown condition under each of the following conditions: (1) a complete loss of computer control capability, (2) a complete station blackout, or (3) a design-basis seismic event.

The NRC staff also concluded that an "actual control room prototype" should be used for test and demonstration purposes. In its SRM dated July 30, 1993, the Commission approved the staff's recommendation. Other potential SMR policy issues include the possible need for requirements on control room staffing during refueling operations, reactor staff who interact with an interconnected manufacturing plant, supervisory staff, shift work, and training.

During pre-application discussions with the NRC staff, SMR designers have indicated that they are evaluating whether the function and task analyses for normal operation and accident management conducted for their SMR designs support control of more than two modules from one control room and support operation with a staffing complement that is less than that currently required by the Commission's regulations. The NRC staff believes that resolution of this issue is required to support the design development, and the staff's review, of design and license applications for most of the SMR designs, including the NGNP. The staff intends to re-assess and revise, as needed, the earlier staff technical positions and plans for resolving the operator staffing issue for SMR designs. Therefore, the issues have been assigned a high importance that should be addressed before submittal of design or license applications of these technology groups. In FY 2010 and FY 2011, the NRC staff will review pre-application white papers and topical reports concerning operator staffing and associated control room design that

it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, discuss the proposed resolutions with human factors and instrument and controls experts, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulatory guidance or staff positions or propose new guidance concerning the operator staffing for an SMR in FY 2012 to support development of the NGNP, integral PWRs, or other SMR designs.

4.2 Operational Programs for Small or Multi-Module Facilities

Scope: Design-specific
Importance: Medium
Issue Paper: FY 2013 or beyond

Policy issues may be identified during the development of operational programs such as inservice inspection and inservice testing programs for SMRs. The unique design of safety-related components, such as the helical steam generators in integral PWRs, may present difficulties and restrictions to the capability to thoroughly conduct the required inspections and tests. The introduction of new technologies and design features may require the development of new operational programs that have not been needed for the current-generation large LWRs or the need to significantly modify current operational programs. On-line refueling and the increased time period between refuelings for certain reactors (from 4 to as many as 30 years between refuelings) may introduce policy issues concerning longer time intervals between periodic inspections and tests. Commission input may be required to determine whether the proposals are acceptable from a policy standpoint..

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. However, the staff believes that resolution for this issue need not occur until after a license application is submitted because it concerns activities that will need to be addressed near the end of an operating license review. Once a license application is received, the NRC staff will review the proposed operational programs for the facility, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and determine the acceptability of the applicant's proposed operational programs. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning these operational programs for the facility in a timeframe consistent with the licensing schedule.

4.3 Installation of Reactor Modules During Operation for Multi-Module Facilities

Scope: Design-specific
Importance: High
Issue Paper: FY 2013 or beyond

The multi-module aspect of certain SMR designs allows modules to be added to the facility while modules that were installed earlier are operating. This type of evolution and possible effects on shared systems and structures could raise policy issues requiring Commission consideration before final decisions regarding the acceptability of a design or issuance of a license are made.

This issue is applicable to license applications for certain integral PWRs. However, the staff believes that resolution for this issue need not occur until after a license application is submitted because it concerns activities that will need to be addressed near the end of an operating license review. Once a license application is received, the NRC staff will review the proposed installation scenario for the facility, consider white papers or topical reports concerning this issue that it receives from the SMR applicant, discuss design-specific proposals to address this matter, and determine the acceptability of the applicant's proposed installation proposal. Should it be necessary, the staff will propose resolutions changes to existing regulatory guidance or new guidance concerning this operational program for the facility in a timeframe consistent with the licensing schedule.

4.4 Industrial Facilities Using Nuclear-Generated Process Heat

Scope: Generic
Importance: High
Issue Paper: FY 2013 or beyond

Besides generating electricity, SMRs provide a possible source of process heat for industrial uses because of their size, high heat production, and capability to be located in remote areas. SMRs are being considered for such industrial uses as producing process heat for chemical plants, refineries, desalinization plants, hydrogen production facilities, and bitumen recovery from oil sands.

The NRC staff has identified potential policy and licensing issues for those facilities used to provide process heat for industrial applications. The close coupling of the nuclear and process facilities raises concerns involving interface requirements and regulatory jurisdiction issues. Effects of the reactor on the commercial product of the industrial facility during normal operation must also be considered. For example, tritium could migrate to a hydrogen production facility and become a byproduct component of the hydrogen product. Resolution of these issues will require interfacing with other government agencies and may require Commission input to determine whether the design and ultimate use of the product is acceptable.

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. However, the staff believes that resolution for this issue need not occur until after a license application is submitted because it concerns site-specific issues associated with the staff's review of an operating license. Once a license application is received, the NRC staff will review the how the nuclear facility is connected to the industrial facility, consider the interrelationship between the staffs of both facility, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and review similar activities with nuclear and non-nuclear facilities. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning the effect of the industrial facility on the nuclear facility in a timeframe consistent with the licensing schedule.

4.5 Security and Safeguards Requirements for SMRs

Scope: Generic
Importance: High
Issue Paper: FY 2011

Traditionally, the approach for security to comply with 10 CFR Part 73, "Physical Protection of Plants and Materials," has largely been one of assessing a plant design and overlaying security provisions (e.g., fences, locked doors, guards) on that design. For SMRs, traditional security provisions could be similar to those for current LWRs. Similarly, material control and accounting (MC&A) safeguards requirements for reactors have been limited to the recordkeeping and other related requirements in 10 CFR 74.19, "Recordkeeping." These would be appropriate and applicable for most of the SMRs. However, SMRs with unique fuel handling requirements may require special licensing requirements for MC&A.

However, since September 11, 2001, it has been recognized that a stronger tie between design and security would be useful so as to integrate the resolution of security issues during the design process. Because many SMRs are still in early developmental stages and the designs are not yet fixed, the designers have a unique opportunity to determine the appropriate design basis threat; develop emergency preparedness; and integrate physical security protection, cyber security protection, and MC&A measures with the design and operational requirements during the design process and during the development of a license applicant's physical security and MC&A programs and systems. Therefore, SMR designers are expected to integrate security into the design and will need to conduct a security assessment to evaluate the level of protection provided, including safeguards aspects of SMR-related fuel cycle and transportation activities.

The small size, reduced number of vital areas, and design approaches that incorporate safety systems underground that characterize the SMR designs have led DOE, SMR designers, and potential SMR operators to raise issues regarding the appropriate number of security staff and size of the protected area. The NRC will need to reevaluate the applicability of the appropriate performance and prescriptive regulatory requirements based on a variety of SMR designs, the design specific source terms to cause radiological sabotage, the enrichment and material forms of special nuclear material, and specific SMR design and license applications. These evaluations will likely require either design or site-specific justifications to support proposed relief from established regulatory requirements or consideration by the Commission before final decisions regarding the acceptability of a design or issuance of a license are made.

The NRC staff believes that resolution of this issue is required to support the design development of the NGNP, integral PWRs, and other SMR designs. Therefore, it has been assigned a high importance that should be addressed before submittal of design or license applications of these technology groups. In FY 2010 and FY 2011, the NRC staff will review pre-application white papers and topical reports concerning safeguards that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, discuss the proposed resolutions with safeguards experts, and consider research and development in this area (both by the domestic and the international community). Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance

concerning safeguards for an SMR in FY 2011 to support development of the NGNP, integral PWRs, or other SMR designs.

4.6 Aircraft Impact Assessments for SMRs

Scope: Design-specific
Importance: High
Issue Paper: FY 2012

On June 12, 2009, the Commission promulgated the Aircraft Impact Rule (74 FR 28112), which requires design and license applicants for new nuclear power reactors to perform a rigorous assessment of their designs to identify design features and functional capabilities that could provide additional inherent protection to avoid or mitigate the effects of an aircraft impact. The applicant is required to identify and incorporate into the design those design features and functional capabilities that avoid or mitigate, to the extent practical and with reduced reliance on operator actions, the effects of the aircraft impact on key safety functions. The applicant is required to show that, with reduced operator actions: (1) the reactor core remains cooled, or the containment remains intact; and (2) spent fuel pool cooling or spent fuel pool integrity is maintained. In its Statement of Considerations for rulemaking, the NRC acknowledged that these requirements may not be applicable to non-LWR designs, or may have to be supplemented by other key functions. When reviewing non-LWR designs, the NRC will evaluate the applicability of the acceptance criteria set forth in the aircraft impact rule and the possible need for other criteria. If necessary, the NRC will issue exemptions and impose supplemental criteria in a design certification or license to be used in the aircraft impact assessment for such non-LWR designs.

Aircraft impact assessments may be needed for future small module design reactors. In addition, aircraft impact issues may have to be addressed for industrial facilities that are using nuclear-generated process heat. Proposed resolutions of this issue for an SMR may require Commission input to determine whether the design approach is in keeping with Commission policy on this issue..

The NRC staff believes that resolution of this issue is required to support the design development of the NGNP, integral PWRs, and other SMR designs. Therefore, it has been assigned a high importance that should be addressed before submittal of design or license applications of these technology groups. In FY 2010 and FY 2011, the NRC staff will review pre-application white papers and topical reports concerning aircraft impact assessments that it receives from DOE and potential SMR applicants, and discuss design-specific proposals to address this matter. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning aircraft impact assessments for SMRs in FY 2011 to support development of the NGNP, integral PWRs, or other SMR designs.

4.7 Offsite Emergency Planning Requirements for SMRs

Scope: Generic
Importance: High
Issue Paper: FY 2012 or beyond

In SECY-93-0092, the NRC staff questioned whether applicants for licenses referencing advanced reactors with passive design safety features should be able to adjust emergency planning zones (EPZs) and requirements. The staff proposed no changes to the existing regulations governing emergency planning for advanced reactor licensees, and stated that it would provide regulatory direction at or before the start of the design certification phase so that emergency planning implications on the design can be addressed. In its SRM dated July 30, 1993, the Commission stated that it was premature to reach a conclusion on emergency planning for advanced reactors and directed the NRC staff to use existing regulatory requirements. However, it instructed the staff to remain open to suggestions to simplify the emergency planning requirements for reactors that are designed with greater safety margins.

Consideration of emergency preparedness by SMR developers is an essential element in the NRC's DID philosophy, which provides that, even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants. However, the smaller size, lower power densities, lower probability of severe accidents, slower accident progression, and smaller offsite consequences per module that characterize SMR designs have led DOE, SMR designers, and potential SMR operators to raise questions regarding the appropriate size of the EPZ, the extent of onsite and offsite emergency planning, and the number of response staff needed. Other topics raised by the industry involve the potential to revise alert and notification requirements and the appropriateness of the protective action requirements in 10 CFR 50.47(b)(10) for SMRs. Although the NRC's current regulations allow for the review of requirements on a case-by-case basis, the Commission may wish to consider such changes for the many designs for which modification is justified. In addition, the applicants requesting certification of their reactor designs may seek finality by having approved changes in offsite emergency planning included as part of the design certification proceeding. Should the applicants propose deviation from NRC requirements, Commission input may be needed to determine whether the proposals are in keeping with Commission policy on this issue..

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. Although resolution of this issue may have a higher importance to an SMR license applicant trying to support its business case at the design certification stage, the staff believes that resolution of this issue may not involve design issues, and therefore, addressing such issues is more appropriate before the COL application stage. A change in the requirements for protective actions and the size of an EPZ is a policy issue that will be of interest to all stakeholders, including the Federal Emergency Management Agency (FEMA) and the public. Any changes to current policies would necessitate appropriate changes to the regulatory requirements and associated guidance documents. This effort would be needed in preparation for COL application reviews. Should it be necessary, the staff will propose changes to existing regulatory requirements and guidance or develop new guidance concerning reduction of offsite emergency preparedness for SMRs in a timeframe consistent with the licensing schedule.

The NRC staff will consider white papers or topical reports proposing to deviate from emergency preparedness requirements that it receives from DOE and potential SMR applicants. During its reviews of COL applications, the staff will discuss site-specific justifications to support proposed deviations, review site-specific proposed emergency preparedness plans, coordinate the reviews with the FEMA, and review similar activities with other nuclear facilities.

5.0 Financial Issues for Small Modular Nuclear Reactors

5.1 Annual Fee for Multi-Module Facilities

Scope: Generic
Importance: Medium
Issue Paper: FY 2011

The 104 power reactors currently licensed to operate have licensed power limits ranging from 1,500 to 3,990 MWt. SMRs are expected to have capacities ranging from 30 to 1,000 MWt. As discussed previously, some of these SMRs may not generate electric power, but instead may be used to generate process heat for industrial applications, such as the production of hydrogen or bitumen recovery from oil sands. Current regulations governing annual fees for power reactors require the same fees from a commercial nuclear reactor designed to produce electrical or heat energy regardless of capacity. SMR developers have identified concerns with this fee structure because of the significant adverse effect on SMR economics.

Although the Commission's regulations allow granting exemptions to the fee requirements if the licensee can justify the reduction in the annual fee, the Commission has issued an advanced notice of proposed rulemaking (ANPR) in March 25, 2009, stating that it is considering whether to amend its regulations to establish a variable annual fee structure for power reactors based on the reactor's licensed power limit contained in operating licenses (including COLs). The ANPR raises issues such as the following:

1. Whether a variable annual fee structure should be based on either the licensed thermal or electric power limits of the power reactor.

2. What the ranges should be for each group or category of reactors if a variable annual fee structure is established.

3. Whether a variable annual fee structure should account for the various configurations made possible by the modular reactors, including single or multiple modules feeding steam to one steam generator or a combination of the application of process heat and electricity production at one facility.

4. Whether and how the fee structure should account for a COL that is issued for a set of modular reactors located at a single site where the licensee can construct, install, and operate each reactor module over a long period of time, depending on the licensee's needs.

The comment period ended on June 8, 2009, and in SECY-09-0137, the NRC staff recommended establishing a working group to analyze options and suggested methodologies

for setting fees for nuclear power reactors, including SMRs. In an SRM dated October 13, 2009, the Commission approved the recommendation. Depending on the working group's recommendation, a proposed amendment to the rule may be presented in a future fee rule.

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. Although resolution of this issue before submittal of a license application may be more important to an SMR license applicant trying to support its business case at the design certification stage, the staff believes that resolution of this issue need not occur until after a licensing application is submitted because it concerns activities that will need to be addressed during an operating license review. However, the likely timing for subsequent Commission papers on this issue provided above is based on the effort associated with the ANPR. The NRC staff will review information submitted as part of the ANPR, including white papers concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and determine whether a proposed amendment to the rule is appropriate. Should it be necessary, the staff will propose changes to the Commission's regulations following the process for processing the ANPR.

5.2 Insurance and Liability for SMRs

Scope: Generic
Importance: Medium
Issue Paper: FY 2011

Section 170 of the Atomic Energy Act (known as the "Price-Anderson Act"), establishes an indemnification and public liability scheme for damages resulting from nuclear power reactor accidents. As discussed previously, SMR configurations include the possibility of using the energy produced by the reactor for process heat in industrial processes, with little or no provision for the actual generation of electricity. Under current law, the maximum public liability for accidents involving non-electric generating SMRs would be much lower than that for comparable electric generating nuclear facilities. If an SMR is not designed and constructed to produce electricity in excess of 100,000 electric kilowatts, it may not be required to participate in the retrospective premium pool established by the Price-Anderson Act, and could be subject to a much lower level of public liability than SMRs designed to produce electricity in excess of 100,000 electric kilowatts.

Therefore, legislation amending the Price-Anderson Act may be necessary to treat non-electricity generating SMRs with no rated electrical generation capacity in a comparable fashion to the electric generating nuclear facilities that are subject to the retrospective premium insurance pool. For example, it may be appropriate for Congress to consider the applicability of the retrospective coverage in the Price-Anderson Act to an SMR with a rated capacity of 300,000 thermal kilowatts rather than 100,000 electric kilowatts. This would clarify that SMRs would be subject to the retrospective insurance pool and higher public liability, thus ensuring that these reactors would be treated the same under the Price-Anderson Act as current commercial nuclear power plants, regardless of those reactors' end-use. Section 140.11(a)(4) of the NRC's regulations tracks the Price-Anderson Act's insurance requirements, including the requirement to maintain retrospective premium insurance, for nuclear reactors designed for the production of electrical energy with a rated capacity of at least 100,000 electrical kilowatts. Accordingly, this regulation would not apply to commercial non-electric generating SMRs. The

current financial protection requirements of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," for all other types of commercial nuclear reactors would apply to non-electric generating SMRs. This is because financial protection requirements for all other types of commercial reactors are based on "thermal power level" rather than electrical kilowatt capacity. See 10 CFR Sections 140.11 and 140.12. The NRC staff will notify the Commission should it conclude that amendments to the Price-Anderson Act or revisions to its regulations may be appropriate.

In addition, in accordance with 10 CFR 50.54(w), separate insurance coverage is required to cover property damage at the site to ensure that the licensee has sufficient funds to stabilize the facility and clean up the site in the event of a nuclear accident. The amount of on-site property insurance required is the lesser of $1.06 billion or whatever amount of insurance is generally available from private sources. This insurance could be a significant cost for an SMR. The amount of insurance required for an SMR may be an issue requiring Commission consideration.

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. Although resolution of this issue before submittal of a license application may be more important to an SMR license applicant trying to support its business case at the design certification stage, the staff believes that resolution of this issue need not occur until after a licensing application is submitted because it concerns activities that will need to be addressed during an operating license review. However, the likely timing for subsequent Commission papers on this issue provided above is based on the need to determine early whether legislation or rulemaking is necessary to address this issue, and how much lead time is necessary to conduct these activities. The NRC staff will consider white papers concerning this issue that it receives from DOE and potential SMR applicants, and determine whether legislation or rulemaking is appropriate to address this issue. Should it be necessary, the staff will propose changes to existing legislation or regulations in a timeframe consistent with the licensing schedule.

5.3 Decommissioning Funding for SMRs

Scope: Generic
Importance: Medium
Issue Paper: FY 2013 or beyond

In SECY-02-0180, the NRC staff questioned whether a non-electric utility may use an alternative method for decommissioning funding for its nuclear power facility, such as partial prepayment. Current NRC regulations allow an applicant several options for funding decommissioning. Non-electric-utility licensees are not allowed to use the sinking fund option exclusively (uniform series of payments). The staff recommended that the NRC require non-electric-utility licensees to use the other options provided in 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," to fund decommissioning costs. At that time, the NRC staff did not recommend that the Commission's regulations be modified to allow additional alternatives for decommissioning funding. In its SRM dated March 31, 2003, the Commission approved the staff's recommendation.

In the same Commission paper, the NRC staff questioned whether a non-LWR applicant could submit design-specific decommissioning cost estimates. The minimum amount of

decommissioning funds required of boiling-water reactors and PWRs is regulated through the minimum decommissioning funds equation in 10 CFR 50.75(c). However, there are no formulas specifically for non-LWR designs. Because the regulations allow the use of a site-specific estimate instead of the amount calculated through the generic formula, the staff stated that it would accept a minimum decommissioning cost estimate specifically for the PBMR or for the GT-MHR if the applicant could technically justify the estimate. For a modular facility, the NRC staff stated that the applicant could submit a standard decommissioning cost estimate based on the decommissioning of one module, which can then be applied multiple times for the facility in question, or (alternatively), a cost estimate based on the decommissioning of multiple modules at a single location. Regardless of the method used, the resulting estimate must include the cost of decommissioning common elements and structures associated with the facility, in addition to the costs of decommissioning each individual module. The NRC staff believes that it is appropriate to accept design-specific decommissioning cost estimates for the potential non-LWRs currently under consideration. In addition, it may be appropriate for the integral PWR designers to submit design-specific decommissioning cost estimates provided adequate justification is provided. The NRC will review each design-specific decommissioning cost estimate submitted on an SMR on a case-by-case basis. Issues concerning the decommissioning costs of an SMR may require Commission consideration.

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. Although resolution of this issue before submittal of a license application may be more important to an SMR license applicant trying to support its business case at the design certification stage, the staff believes that resolution of this issue need not occur until after a licensing application is submitted because it concerns activities that will need to be addressed during an operating license review. Once a license application is received, the NRC staff will review the associated design-specific decommissioning cost estimate, consider white papers concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and determine whether the estimate is acceptable in light of current regulations and regulatory guidance. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning decommissioning costs for SMRs in a timeframe consistent with the licensing schedule.

Descriptions of Small Modular Nuclear Reactor Designs

The following are design descriptions of the small modular integral light-water reactor (LWR), high-temperature gas-cooled reactor, and sodium-cooled fast reactor (SFR) designs that have been under development by nuclear reactor designers, who have notified the U.S. Nuclear Regulatory Commission (NRC) that that they may submit design and license applications for some of their designs to the NRC as early as fiscal year (FY) 2011.

Next Generation Nuclear Plant

In Subtitle C of the Energy Policy Act of 2005 (EPAct), Section 641 states that the Secretary of Energy shall establish the Next Generation Nuclear Plant (NGNP) project, which will consist of constructing, licensing, and operating a prototype nuclear plant that can be used to generate electricity, hydrogen, or both. As defined by the EPAct, the NGNP will be a full-scale prototype plant that will be reliable, safe, proliferation resistant, and economical and will demonstrate the commercial potential of the design and associated technologies. Although the prototype NGNP is planned to be a single unit, issues regarding multi-module operation could be applicable to future commercial NGNP applications. The mission of the NGNP includes providing high-temperature process heat for the chemical industry, refining petroleum, extracting oil from shale and tar deposits as an alternative to natural gas, producing hydrogen, and serving as a central electric power station. To meet this mission, the Department of Energy has concluded that the NGNP should be a gas-cooled, very-high-temperature reactor, and could be considering designs such as the Pebble Bed Modular Reactor (PBMR), the Gas-Turbine Modular Helium Reactor (GT-MHR), and AREVA's New Technology Advanced Reactor Energy System (ANTARES). These designs have the potential to produce the high-temperature heat needed to support the mission of the NGNP while relying on inherent characteristics and passive safety features to mitigate design-basis accidents (DBAs). The following describes each of these designs:

- **Pebble Bed Modular Reactor**

  The Pebble Bed Modular Reactor (PBMR) is a 400 megawatt-thermal (MWt) modular high-temperature helium-cooled reactor under development by PBMR (Pty.) Ltd. Its baseline configuration is for use as an electric power plant with a power output ranging from 165 megawatt-electric (MWe) (i.e., one reactor module) to 1320 MWe (i.e., eight reactor modules). The PBMR module consists of a graphite-moderated, helium-cooled reactor and a direct closed-cycle turbine-driven generator. The 450,000 fuel pebbles that comprise the core are billiard-ball-sized graphitic spheres containing fuel kernels composed of low-enriched (9 percent) uranium dioxide (UO_2) coated with a fission-product-retaining tri-structural isotropic (TRISO) coating. The PBMR reactor core and fuel are based on the high-temperature gas-cooled reactor technology that was originally developed in Germany.

- **Gas-Turbine Modular Helium Reactor**

  The Gas-Turbine Modular Helium Reactor (GT-MHR) is a 600 MWt (285 MWe) modular high-temperature helium-cooled reactor (MHTGR) under development by General Atomics that consists of a graphite-moderated, helium-cooled reactor and a direct closed-cycle turbine-driven generator. The fuel is in the form of graphitic cylindrical fuel compacts containing fuel kernels composed of low-enriched (10 – 19.9 percent) UCO coated with a fission-product-retaining TRISO coating. The fuel compacts are inserted into hexagonal prismatic graphite blocks. The GT-MHR design is based on the Fort St. Vrain, and later MHTGR, designs developed by General Atomics.

- **AREVA's New Technology Advanced Reactor Energy System**

  The AREVA's New Technology Advanced Reactor Energy System (ANTARES) is a 600 MWt (285 MWe) modular high-temperature helium-cooled reactor under development by AREVA that consists of a graphite-moderated, helium-cooled reactor and indirect-cycle gas- and steam-turbines using intermediate heat exchangers. The fuel is in the form of graphitic cylindrical fuel compacts containing fuel kernels composed of low-enriched (10 – 19.9 percent) UO_2 coated with a fission-product-retaining TRISO coating. The fuel compacts are inserted into hexagonal prismatic graphite blocks.

Super-Safe, Small and Simple Reactor

The Super-Safe, Small and Simple Reactor (4S) reactor is a small, 30 MWt (10 MWe) pool-type SFR, designed by the Toshiba Corporation, that is intended for use in remote locations where it could operate for up to 30-years without the need for refueling. The 4S reactor is designed to rely on inherent safety characteristics and passive features to achieve all safety functions for all licensing basis events. The reference 4S reactor design produces 10 MWe, although both larger and smaller 4S reactor designs are also proposed. Fuel inside steel-clad rods is composed of a uranium-zirconium alloy at enrichments of 17 and 19 percent U-235. For deployment in the United States, the major components of the 4S reactor, including the fuel and reactor vessel, would be fabricated at a factory, shipped to the intended site, and assembled and installed underground in a below-grade civil structure.

Power Reactor Inherently Safe Module

The Power Reactor Inherently Safe Module (PRISM) reactor is a modular, pool-type SFR design first developed by the General Electric Company. The standard plant design for the PRISM consists of three identical power blocks with a total electrical output rating of 1395 MWe. Each power block comprises three reactor modules, each with an individual thermal rating of 471 MWt (155 MWe). Each module is located in its own below-grade silo and is connected to its own intermediate heat transport system and steam generator system. The reactor core consists of a metallic-type fuel rod composed of a ternary alloy of uranium-plutonium-zirconium clad in steel. The design includes passive reactor shutdown and passive decay heat removal features.

International Reactor Innovative and Secure

The International Reactor Innovative and Secure (IRIS) is a 1000 MWt (about 335 MWe) modular pressurized-water reactor (PWR) design with an integral configuration, under development by an international consortium of more than 30 organizations from nine countries, led by Westinghouse Electric Company. All primary system components (pumps, steam generators, pressurizer, and control rod drive mechanisms) are inside the reactor vessel. The reactor uses traditional PWR fuel rods (less than 5-percent enrichment) arrayed in 17x17 fuel bundles. A power station could be built with one or more modules. IRIS has an extended core life of up to 48 months. IRIS is designed to rely on passive safety features to mitigate design basis accidents. Its design for electric power generation has progressed to the integrated testing phase, and it is currently in the final design and development phase.

NuScale Power Reactor

The NuScale Power Reactor is a 150 MWt (45 MWe) natural circulation PWR design that consists of a self-contained assembly with the reactor core and steam generators located in a common reactor vessel. The reactor uses approximately one-half-height PWR fuel rods (less than 5-percent enrichment) arrayed in 17x17 bundles. The NuScale light-water reactor design employs a non-traditional, small containment for each module that operates in a common, large pool of water. Electrical power conversion involves the use of steam generators and a steam turbine-generator. NuScale Power, Inc., plans to submit a design certification application for a 12-module facility. These modular units would be manufactured at a single centralized facility; transported by rail, road, and/or ship; and installed as a series of self-contained units, each with a 24-month refueling cycle. The design is being developed by NuScale Power, Inc.

mPower Reactor

The mPower Reactor is a 400 MWt (125 MWe) PWR module that consists of a self-contained assembly with the reactor core, reactor coolant pumps, and steam generators located in the reactor vessel. The mPower reactor, under development by the Babcock & Wilcox Company, employs control rods but no soluble boron for normal reactivity control. The reactor uses approximately one-half-height PWR fuel rods (less than 5-percent enrichment) arrayed in 17x17 bundles. The reactor module uses a once-through steam generator and plans on a 5-year fuel cycle. The designer is still in the process of determining whether two modules will feed one turbine-generator through a common steam header to produce a total of 250 MWe.

Key Documents Concerning Policy, Licensing, and Key Technical Issues For Small Modular Reactors[4]

The Atomic Energy Act of 1954, as amended (P.L. 83-703), dated August 30, 1954. (ADAMS Accession No. ML022200075)

SECY-88-0202, "Standardization of Advanced Reactor Designs," U.S. Nuclear Regulatory Commission, July 15, 1988. (ADAMS Accession No. ML051740706)

SECY-88-0203, "Key Licensing Issues Associated with DOE-Sponsored Advanced Reactors," U.S. Nuclear Regulatory Commission, July 15, 1988. (ADAMS Accession No. ML051830035)

Staff Requirements Memorandum (SRM), "Commission Action on the Key Licensing and Standardization Issues Associated with the DOE Advanced Reactor Concepts (SECY-88-202 and SECY-88-203), U.S. Nuclear Regulatory Commission, September 19, 1988. (ADAMS Accession No. ML010650233)

SRM Regarding "SECY-88-202 - Standardization of Advanced Reactor Designs and SECY-88-203 - Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs," U.S. Nuclear Regulatory Commission, November 14, 1988. (ADAMS Accession No. ML010940239)

SECY-91-0202, "Departures from Current Regulatory Requirements in Conducting Advanced Reactor Reviews," U.S. Nuclear Regulatory Commission, July 2, 1991. (ADAMS Accession No. ML051740732)

SECY-93-0092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, April 8, 1993. (ADAMS Accession No. ML040210725)

SRM, "SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, July 30, 1993. (ADAMS Accession No. ML003760774)

NUREG-1368, "Pre-Application Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," U.S. Nuclear Regulatory Commission, February 1994. (ADAMS Accession No. ML063410561)

SECY-95-035, "Reassessment of Fee Billing Practices and Fee Policy for Office of Nuclear Regulatory Research (RES) Activities Associated with Design Certification (DC) Applications," U.S. Nuclear Regulatory Commission, February 15, 1995. (ADAMS Accession No. ML023230188)

[4] All documents referenced in this attachment are available in the Agencywide Documents Access and Management System (ADAMS) on the NRC's Web site (http://www.nrc.gov) under the accession numbers provided.

SRM, "SECY-95-035 - Reassessment of Fee Billing Practices and Fee Policy for Office of Nuclear Regulatory Research (RES) Activities Associated with Design Certification (DC) Applications," U.S. Nuclear Regulatory Commission, March 10, 1995. (ADAMS Accession No. ML003756447)

NUREG-1338, "Pre-Application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) - Draft Copy of the Final Report," U.S. Nuclear Regulatory Commission, December 1995. (ADAMS Accession No. ML052780519)

"The Role of Defense-In-Depth in a Risk-Informed Regulatory System," Letter from the Advisory Committee on Reactor Safeguards, May 19, 1999. (ADAMS Accession No. ML091280427)

"Regulatory Issues Related to the Pebble Bed Modular Reactor (PBMR)," Letter from Exelon Generation to NRC, May 10, 2001. (ADAMS Accession No. ML011420393)

SECY-01-0207, "Legal and Financial Issues Related to Exelon's Pebble Bed Modular Reactor (PBMR)," U.S. Nuclear Regulatory Commission, November 10, 2001. (ADAMS Accession No. ML012850139)

SRM, "Staff Requirements - SECY-01-207 - Legal and Financial Issues Related to Exelon's Pebble Bed Modular Reactor (PBMR)," Letter from the Advisory Committee on Reactor Safeguards, January 14, 2002 (ADAMS Accession No. ML020150058)

"Industry White Paper - Integrated Approach to Modular Plant Licensing, Nuclear Energy Institute, June 17, 2002. (ADAMS Accession No. ML021970596)

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POLICY ISSUE
INFORMATION

April 7, 2010 SECY-10-0042

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: REACTOR OVERSIGHT PROCESS SELF-ASSESSMENT FOR CALENDAR YEAR 2009

PURPOSE:

The purpose of this paper is to present the results of the staff's annual self-assessment of the Reactor Oversight Process (ROP) for calendar year (CY) 2009.

SUMMARY:

The results of the CY 2009 self-assessment indicate that the ROP met its program goals and achieved its intended outcomes. The staff of the U.S. Nuclear Regulatory Commission (NRC) found that the ROP met the agency's strategic goals of ensuring safety and security through objective, risk-informed, understandable, and predictable oversight. The staff implemented several ROP improvements in CY 2009 to address issues raised by the Commission and obtained through feedback from internal and external stakeholders.

The staff continues to improve existing performance indicators (PIs) and explore potential new indicators to ensure that the PI program provides meaningful input to the ROP. The NRC independently verified through its inspection program that plants were operated safely and securely, and the NRC ensured that sites remained staffed with knowledgeable and experienced inspectors. The significance determination process (SDP) remained an effective tool for determining the safety and security significance of identified performance issues in a

CONTACT: Ronald K. Frahm, Jr., NRR/DIRS
(301) 415-2986

timely manner. The assessment program provided for regulatory oversight in identifying licensee performance issues and determining appropriate regulatory response. The staff will continue to solicit input from the NRC's internal and external stakeholders and further improve the ROP based on stakeholder feedback and lessons learned.

BACKGROUND:

The staff performed the CY 2009 self-assessment in accordance with Inspection Manual Chapter (IMC) 0307, "Reactor Oversight Process Self-Assessment Program." The ROP self-assessment program uses program evaluations and performance metrics to evaluate the overall effectiveness of the ROP in meeting its preestablished goals and intended outcomes.

The ROP includes the four specific program goals of being objective, risk informed, understandable, and predictable, as well as the applicable organizational excellence objectives (openness and effectiveness) from the NRC's Strategic Plan for Fiscal Years (FYs) 2008–2013. Each of these ROP goals supports the NRC's mission and characterizes the manner in which the agency achieves its strategic goals of safety and security. The intended outcomes of the ROP, which help form its basis and are incorporated into the various ROP processes, include the following:

- appropriately monitoring and assessing licensee performance
- identifying performance issues through NRC inspection and licensee PIs
- determining the significance of identified performance issues
- adjusting resources to focus on significant performance issues
- evaluating the adequacy of corrective actions for performance issues
- taking necessary regulatory actions for significant performance issues
- communicating inspection and assessment results to stakeholders
- making program improvements based on stakeholder feedback and lessons learned

DISCUSSION:

During the tenth year of ROP implementation (CY 2009), the staff conducted numerous activities and obtained data from many diverse sources to ensure that it performed a comprehensive and robust self-assessment. Data sources included the ROP performance metrics described in IMC 0307, feedback received from internal and external stakeholders, and direction and insight contained in several Commission staff requirements memoranda (SRM). The staff analyzed the information from these various sources to gain insights regarding ROP effectiveness and potential areas for improvement. The scope of the staff's ROP self-assessment included the key ROP program areas, ROP communication activities, independent evaluations, ROP resources, and resident inspector (RI) demographics and staffing. As noted in the pertinent sections of this paper, the staff has also included several enclosures with additional detail to support its self-assessment and conclusions.

ROP Program Area Evaluations

The staff evaluated each of the four key program areas of the ROP: the PI program, inspection program, SDP, and assessment program. The results are summarized below and are discussed in more detail in Enclosure 1. In addition, the annual ROP performance metric report, available through the Agencywide Documents Access and Management System (ADAMS),

provides the data and staff analysis for each program area metric (ADAMS Accession No. ML100540037).

PI Program—The staff continued to improve existing PIs, reinforce the guidance and expectations governing the reporting of PI data, and explore potential new indicators in CY 2009 to ensure that the PI program provides meaningful inputs to the ROP. The staff met all eight of the PI metrics for CY 2009. The external survey of stakeholders generally found that the PI program gave an objective indication of declining safety performance, contributed useful information in risk-significant areas, was clearly defined and understandable, and provided an appropriate overlap with the inspection program. During CY 2009, the staff improved the effectiveness of the mitigating system performance index (MSPI) as a result of the lessons learned review. The staff also provided safety system functional failure (SSFF) training to the regional inspectors to enhance their awareness of the reporting requirements and governing guidance. The staff evaluated PIs in current use by the industry for their potential efficacy within the ROP. The staff also reviewed PIs already in use by the United States and international nuclear power industries (i.e., non-ROP PIs) for potential applicability to the ROP PI program. The results of the staff's review were documented in a white paper that was shared with external stakeholders. The staff will continue to refine existing PIs and engage stakeholders in a discussion of potential new PIs for ROP implementation.

Inspection Program—NRC inspectors independently verified that plants were operated safely and securely. All inspection program metrics were met, including the completion of the required baseline inspection program for CY 2009. The staff made changes to selected ROP inspection procedures (IPs) based on completion of the third ROP realignment. The staff continued to use operating experience (OpE) information in the baseline inspection program, including the OpE Smart Sample process and several others, and is considering initiatives to further integrate OpE into inspection program processes and activities. An NRC senior-level management working group also developed strategies and initiated actions to address challenges to RI retention issues and reported these enhancements to the Commission. External survey responses were favorable on the quality of inspection reports and the adequacy of the inspection program's coverage of areas important to safety and security.

SDP—The SDP continues to be an effective tool for determining the safety and security significance of identified performance issues. The staff met the SDP timeliness metric for the fourth consecutive year and also met all other SDP metrics. The staff issued several SDP guidance documents in CY 2009, including the new SDP Appendix L for alternative mitigation strategies (B.5.b) and the revised baseline security SDP. The staff continues to develop analytical tools for low-power and shutdown applications, with four models available for use, two being developed, and one planned. A team, comprising staff members from the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Regulatory Research (RES), and the Regions, implemented a partnering initiative to review the NRC risk tools to identify areas for enhancement. The responses to the external survey indicated that, overall, the stakeholders thought the SDP resulted in the appropriate regulatory response, although they suggested areas for improvement. The staff plans additional SDP development and training for CY 2010.

Assessment Program—Implementation of the NRC's assessment program ensured that staff and licensees focused on addressing performance issues. The staff revised IMC 0305, "Operating Reactor Assessment Program," to improve usability, incorporate guidance on traditional enforcement, clarify safety culture concepts, incorporate operating experience, and

respond to stakeholder feedback. The staff also enhanced internal and external communications of plant assessment results, including a revision to the action matrix public Web site to provide a more current status of plant assessment, rather than a purely retrospective look at the previous quarter's data. During CY 2009, the staff observed a decline in the number of plants in the degraded cornerstone (Column 3) and the multiple/repetitive degraded cornerstone (Column 4) of the action matrix. At the Commission's request, the staff provided the plans and schedules for satisfying the criteria to return two plants to normal NRC monitoring efforts in SECY-09-0121, "Status of the Deviation from the Reactor Oversight Process Action Matrix for Davis-Besse Nuclear Power Station and Indian Point Energy Center," dated August 24, 2009. As of the end of CY 2009, the staff had closed out both the Davis-Besse and Indian Point deviations, and there are some deviations from the action matrix that are in process in late CY 2009 and CY 2010.

The agency met seven of the eight assessment metrics for CY 2009, including all timeliness goals. In the 2009 external ROP survey, the perception of the assessment program was generally positive. However, the NRC did not meet one metric as a result of negative feedback on safety culture in the external survey from the industry. The staff is aware of the industry's concern with the process for determining substantive cross-cutting issues and will continue to consider industry proposals as noted below. The staff implemented several changes to ROP guidance in CY 2009, including detailed guidance for performing an independent safety culture assessment. The staff also developed training for regional staff on the NRC's ongoing safety culture activities related to the ROP. The staff leveraged ongoing efforts initiated by the Deputy Regional Administrators to improve the reliability of ROP implementation, including the substantive cross-cutting issue process. In addition, the Nuclear Energy Institute (NEI) proposed an alternative industry-owned safety culture oversight process, which the NRC staff is currently observing to become familiar with the initiative and to evaluate associated tools that could possibly be leveraged to gain efficiencies in the ROP.

ROP Communication Activities

The staff continued to emphasize stakeholder involvement and open communications regarding the ROP throughout CY 2009. The staff used a variety of communication methods to ensure that all stakeholders could access ROP information and could both participate in the process and provide feedback. As discussed below, the staff sought and implemented improvements to the ROP, based on feedback and insights from all stakeholders.

Internal Stakeholder Interface—NRR staff and staff from the Office of Nuclear Security and Incident Response (NSIR) continued to conduct monthly conference calls with regional management and staff to discuss current issues associated with the ROP. The staff also met periodically with regional managers to discuss more complex ROP issues. In addition, the staff participated in each region's inspector counterpart meeting to provide specific training and to gather regional feedback on ROP implementation. The staff also conducted periodic counterpart calls among headquarters and regional staff on a variety of topics such as materials engineering, fire protection, and security topics. These counterpart calls ensured that regional staff remains cognizant of emerging technical and policy issues while headquarters staff maintained awareness of plant safety and security issues.

The NRC staff effectively used the ROP feedback process to identify concerns or issues and recommend and implement improvements related to ROP policies, procedures, or guidance.

For CY 2010, the NRR staff plans to improve the communication of information related to this process to internal stakeholders by posting information on the NRC SharePoint portal. The NRC staff frequently updated the ROP Digital City Web site to include recent and useful information for internal stakeholders. The NRC staff continued to issue the inspector newsletter on a quarterly basis to share value-added inspection findings, best practices, inspection guidance, and regulatory issues of interest to inspectors and staff implementing the ROP. The inspector newsletter is also represented as a community of practice on the NRC's knowledge management Web site, which provides a place for inspectors to seek and discuss information that appeared in newsletter articles. The staff continued to improve the initial and continuing inspector training programs to develop and maintain well-qualified, competent inspectors, as discussed in Enclosure 1.

External Stakeholder Interface—The staff continued to conduct monthly public working-level meetings with NEI, the industry, and interested stakeholders to discuss the status of ongoing refinements to the ROP. The staff also held public events in the vicinity of each operating reactor to discuss the results of the NRC's assessment of the licensee's performance and provide an opportunity to engage interested stakeholders on the NRC's role in ensuring safe and secure plant operations. Additionally, regional staff participated in various local community information meetings involving licensed facilities and conducted outreach activities with other federal agencies, state and local officials and private organizations. The staff also worked with external stakeholders on the development of the Force-on-Force (FOF) inspection and SDP enhancements. The staff published the Annual Report to Congress on the Security Inspection Program in July 2009 to continue to communicate information and results related to the security cornerstone. The staff also sponsored a breakout session on ROP initiatives at the Regulatory Information Conference in March 2009 and discussed additional ROP topics during the regional breakout sessions. The staff maintained and enhanced the NRC's Web pages to communicate current ROP-related information and results. For example, based on stakeholder feedback, the staff revised the Web page for the action matrix summary to provide more current information on the level of regulatory oversight being applied to all operating reactor units.

Stakeholder Survey Results— On September 25, 2009, the staff issued its external survey in a *Federal Register* notice (FRN) to evaluate ROP effectiveness and gather stakeholder insights. The survey requested responses to 21 specific questions corresponding to ROP performance metrics as defined in IMC 0307. To maximize awareness of the survey's availability, the staff also (1) mailed more than 500 surveys directly to stakeholders, (2) placed a direct link to the survey information on both the ROP Web page and the "Documents for Comment" page of the NRC's external Web site, and (3) issued a press release. The staff did not conduct an internal survey in CY 2009, consistent with the biennial frequency prescribed by IMC 0307.

The NRC received five responses to the FRN from the individuals or organizations listed below. These responses are available in ADAMS, under the accession numbers in parentheses following the respondent's name:

- Southern Nuclear (ML093140305)
- Nuclear Energy Institute (ML093140556)
- Region IV Utility Group (ML093140557)
- Strategic Teaming and Resource Sharing (ML093140558)
- Respondent from Wolf Creek Nuclear Operating Corporation (ML093290157)

The responses from the survey of external stakeholders were all from utility representatives, and the number of responses continued to decline. The agency received only 5 responses for the CY 2009 survey, down from the 7 responses for the CY 2007 survey, 16 in CY 2006, and 21 in CY 2005. For the first time since ROP implementation, the agency received no responses from interested public representatives or State or local agencies. As a result of the declining number and breadth of survey participants, the staff plans to reconsider the content and frequency of the ROP surveys or potentially explore alternate venues to obtain stakeholder feedback. The responses were generally positive, but some noted concerns and areas for improvement. The staff's analysis of the survey responses appears in the applicable portions of the program area evaluations in Enclosure 1, as well as in the annual ROP performance metrics report. In addition, as for previous external surveys and as formalized in IMC 0307, the staff will prepare a consolidated response to the CY 2009 external survey to more specifically address the comments received.

ROP Performance Metrics and Independent Evaluations

ROP Performance Metrics—Based on the NRC staff's review, all but one of the 45 performance metrics for the ROP met the established criteria as defined in Appendix A to IMC 0307, "Reactor Oversight Process Self-Assessment Program." All 8 metrics in the PI program area, all 7 metrics in the inspection program area, all 6 metrics in the SDP area, 7 of the 8 metrics in the assessment program area, and all 16 overall ROP program metrics met the established criteria. The NRC did not meet the one metric as a result of negative feedback on safety culture in the external survey from the industry. The staff is aware of the industry's concern and will continue to consider industry proposals as previously noted. The staff further discusses the performance metrics in the program area evaluations in Enclosure 1, as well as in the annual performance metric report (ADAMS Accession No. ML100540037).

Independent Evaluations—In addition to the ROP self-assessment program, the staff has received several independent evaluations of ROP effectiveness in the past few years. These evaluations generally provided favorable results, but they also suggested potential areas of improvement. Most recently, the staff hired FocalPoint Consulting Group to perform an independent evaluation of the reactor oversight and incident response programs in late 2008 and develop recommendations for strengthening program performance. While FocalPoint found the programs to be effective in accomplishing their objectives of providing reactor oversight and incident response, it provided a number of findings and recommendations for the staff's consideration. In 2009, the staff reviewed the report and developed a comprehensive table of the staff's response and status for each of the recommendations, many of which the staff had already identified and was implementing. Greater detail on the independent evaluations of the ROP along with the staff's response and resultant program improvements appear on the ROP Web page entitled "ROP Program Evaluations and Stakeholder Feedback."

Regulatory Impact—The staff also received and evaluated feedback from licensees as part of the regulatory impact process. This process, established in 1991, followed the Commission's direction to develop a method for obtaining feedback from licensees and reporting the feedback to the Commission. Over the past year, the staff received and compiled feedback from 95 site visits to 43 reactor sites (68 units) across all four regions. These visits resulted in 178 distinct comments that fell into two main categories—formal communications with licensees and inspector performance. Of the comments compiled, 92 percent were favorable and 8 percent were unfavorable. The number and distribution of comments and the favorable percentage

were similar to previous years. Enclosure 2 provides a summary of the feedback received and the staff's evaluation and actions to address the noted concerns.

Industry Performance Trends—The NRC collects and monitors industrywide data to assess whether the nuclear industry, as a whole, is maintaining the safety performance of operating plants. The NRC also uses these industry indicators as feedback for improving the ROP. The staff is reporting the FY 2009 results of the Industry Trends Program to the Commission in an annual paper that complements this paper. The results of the Industry Trends Program will also be reviewed at the Agency Action Review Meeting.

ROP Resources

Overall staff effort in FY 2009, as reflected in expended hours, increased by 1.4 percent, compared with FY 2008. Baseline inspection hours increased in 2009 primarily as a result of increased effort in performing IP 71152, "Identification and Resolution of Problems," and IP 71130.03, "Contingency Response—Force-on-Force Testing." Although more of these inspections were performed in FY 2009 than in FY 2008, the staff will consider this apparent increase in inspection hours during the next ROP realignment of inspection resources. The hours charged to other baseline procedures remained relatively unchanged.

Total ROP effort during the past three years has remained relatively stable at approximately 6,300 hours per site and is consistent with the budgeted resources. The small annual variances are likely the result of (1) baseline inspection realignment with attendant changes in inspection cycle frequency, (2) year-to-year implementation variations in the first, second and third year of the inspection cycle for procedures with multi-year frequencies, and (3) the annual variation in plant-specific inspections in response to licensee performance and emerging generic safety issues. Enclosure 3 discusses ROP resources in greater detail.

Resident Inspector Demographics and Site Staffing

As directed in an SRM dated April 8, 1998, the staff developed measures to monitor and trend RI demographics and report the results to the Commission annually. The staff later developed a site staffing metric that is included with the annual analysis. The staff concluded that sites continue to be staffed with knowledgeable and experienced RIs and senior resident inspectors (SRIs). Staff turnover within the NRC, whether caused by promotion, reassignment, retirement, or resignation, is an ongoing process from which the RI program is not insulated. The turnover in the RI ranks over the last several years resulted in a decline of onsite inspection experience, but the turnover rates in both RI and SRI ranks have improved from 2007 through 2009. Nonetheless, the NRC has initiated several actions to ensure an experienced and stable RI and SRI program. The staff reported these enhancements to the Commission in SECY-09-0050, "Actions to Enhance Relocation and Retention for Employees," dated March 30, 2009. The staff plans to continue closely monitoring resident demographics and site staffing in 2010. In accordance with the SRM dated June 26, 2009, the staff will report on the effectiveness of the relocation and retention enhancements for SRIs and RIs in a separate paper to the Commission in CY 2011. Enclosure 4 provides detailed analyses of the 2009 RI demographics and site staffing.

COMMITMENTS:

Prior Commitments—The staff made eight commitments in last year's ROP self-assessment to improve the efficiency and effectiveness of the ROP. The following summarizes the actions taken by the staff to address these eight commitments:

(1) The staff continued to implement improvement initiatives based on its MSPI Lessons learned review and provided training on the SSFF PI to the inspection staff, as described in Enclosure 1.

(2) The staff revised program guidance to better integrate OpE into the ROP assessment process; and it continues to emphasize the use of OpE and plans to further integrate this emphasis into the inspection program in CY 2010, as described in Enclosure 1.

(3) The staff provided recommendations in a separate paper to the Commission detailing potential improvements to the relocation and retention practices for RI and SRI staff, as described in Enclosure 4.

(4) The staff initiated the development of additional SDP training to ensure that inspectors remain efficient and effective in determining the safety and security significance of identified performance issues and will continue these efforts in CY 2010, as described in Enclosure 1.

(5) The staff developed and implemented several models for low-power and shutdown situations for use in the SDP, and it plans additional models, as described in Enclosure 1.

(6) The staff revised program guidance to better integrate traditional enforcement outcomes into the assessment process, as described in Enclosure 1.

(7) The staff will revise program guidance, as necessary, to better align with the Commission's safety culture policy statement, once it has been completed, as described in Enclosure 1. Since a final safety culture policy statement was not established in CY 2009, the staff is carrying this commitment into CY 2010.

(8) The staff explored ways to use cross-regional experience to further improve the implementation of the substantive cross-cutting issue guidance and other areas of the ROP, as described in Enclosure 1.

New Commitments—As described in this paper, the staff plans the following five significant actions or ongoing activities to improve the efficiency and effectiveness of the ROP in CY 2010:

(1) The staff will develop a framework for evaluating the efficacy of potential new PIs for use in the ROP.

(2) The staff will continue to emphasize the availability and use of OpE in the inspection program and plans to further integrate this emphasis into the inspection guidance.

(3) The staff will conduct additional SDP training based on input from the partnering initiative, which provided valuable insights regarding areas where training was lacking or can be improved.

(4) In accordance with SRM M100112, "Briefing On Office Of Nuclear Security and Incident Response-Programs, Performance, And Future Plans", dated February 12, 2010, the staff will report back to the Commission on how the proposed enhancements to the FOF physical protection SDP would alter the CY 2009 FOF exercise findings.

(5) The staff will revise ROP program guidance, as necessary, to align with the Commission's safety culture policy statement, once it has been completed.

The staff will include the status of these commitments and the other program improvements noted in this paper in the CY 2010 ROP self-assessment.

CONCLUSIONS:

The self-assessment results for CY 2009 indicate that the ROP provided effective oversight by meeting the program goals and achieving its intended outcomes. The ROP was successful in being objective, risk informed, understandable, and predictable. The ROP also ensured openness and effectiveness in support of the agency's mission and its strategic goals of safety and security. The NRC appropriately monitored operating nuclear power plant activities and focused agency resources on performance issues in CY 2009, and plants continued to receive a level of oversight commensurate with their performance. The ROP has developed into a mature oversight process over the past 10 years; however, the staff continues to refine it in response to emerging issues, lessons learned, and suggested improvements from internal and external stakeholders.

RESOURCES:

NRC headquarters and regional resources are needed to conduct the periodic assessment and realignment of ROP inspection procedures, ROP annual program assessment, mid-cycle and end-of-cycle licensee performance assessment; to revise and maintain the NRC Inspection Manual; and to perform all ROP management and oversight activities. The staff estimates that 56.5 full-time equivalent (FTE) staff members and $875,000 will be needed for FY 2010 to conduct these NRR-funded activities. In FY 2011, it will require 64.3 FTE and $939,000.[1]

In addition, NSIR estimates that it will require approximately 43.4 FTE for FY 2010 and 42.6 FTE in FY 2011 for its ROP inspection and support activities and for licensee performance assessments. RES estimates that it will require approximately 1.9 FTE and $985,000 for FY 2010 and 1.8 FTE and $908,000 for FY 2011 for its ROP assistance programs. NSIR and RES budget and perform their portion of the work separate from the NRR effort. The staff does not anticipate that it will require any resources beyond those already included in the current budget requests for FY 2010 and FY 2011 for these activities. The staff will address resource

[1] The FY 2011 resource requirements include 8 FTE for inspector development as part of the Resident Inspector recruitment and retention initiative. Other ROP management and oversight activities in FY 2011 remain stable at 56.3 FTE and comparable to FY 2010 requirements.

requirements beyond FY 2011 during the planning, budgeting, and performance management process of the respective year.

COORDINATION:

The Office of the General Counsel has reviewed this Commission paper and has no legal objection. The Office of the Chief Financial Officer has reviewed this Commission paper and determined that there is no financial impact.

/RA by Bruce S. Mallett for/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Reactor Oversight Process Program Area Evaluations
2. Regulatory Impact Summary
3. Reactor Oversight Process Resources
4. Resident Inspector Demographics

Reactor Oversight Process Program Area Evaluations

In accordance with Inspection Manual Chapter (IMC) 0307, "Reactor Oversight Process Self-Assessment Program," the staff of the U.S. Nuclear Regulatory Commission (NRC) performed program evaluations in each of the four key program areas of the Reactor Oversight Process (ROP), including performance indicators (PIs), inspection, significance determination process (SDP), and assessment. The staff used self-assessment metrics, feedback from internal and external stakeholders, and other information to gain insights into the effectiveness of the ROP in meeting its goals and intended outcomes. Based on the metric results, stakeholder comments, and other lessons learned through ongoing program monitoring, the staff identified certain issues and actions in each of the four key program areas, as described below. The annual ROP performance metric report provides the data and staff analysis for each of the program area metrics (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100540037).

Performance Indicator Program

During calendar year (CY) 2009, the staff continued to look for ways to improve the effectiveness of the PI program. The staff reinforced the guidance and expectations governing the gathering and submittal of data for existing PIs through the frequently asked question (FAQ) process. For example, the staff reinforced the reporting requirements and governing guidance for the safety system functional failure (SSFF) PI to the NRC inspection staff and industry. In December 2009, the staff provided training on the SSFF PI to the regional inspectors at the semi-annual counterpart meetings, as it committed to do in the CY 2008 self-assessment. As evidenced by audience feedback, the SSFF training was generally well received and was noted for its direct applicability to both the inspection and PI programs.

The staff continually looks for ways to modify and improve existing PIs to ensure their effectiveness. As it committed to do in last year's self-assessment, the staff improved the mitigating system performance index (MSPI) as a result of the recently completed lessons learned review. This review generated several staff white papers, PI guidance changes, and other activities to improve the effectiveness of the MSPI. Two staff MSPI white papers have been resolved, and two others will be resolved by end of CY 2010. The two issues that were resolved concerned properly accounting for rounding errors when computing the final MSPI values and changing the MSPI planned train unavailability baseline. The two staff white papers not yet resolved concern monitoring emergency diesel generator (EDG) fuel oil transfer pumps as part of the MSPI and revising the component failure mode definitions for EDGs. The staff has formally developed additional initiatives regarding certain component boundaries and failure mode definitions. The NRC will continue to discuss these initiatives, along with any future efforts for MSPI improvement, in the ROP Working Group monthly public meetings.

In addition, the staff reviews and assesses the effectiveness of the security PI on an annual basis as part of its self-assessment. Based on this review, the staff discussed its self-assessment with stakeholders from the NRC, industry, state governments, and the public. The stakeholders discussed the publication of the new requirements of 10 CFR Part 26 and 73 and resultant changes to the baseline inspection program. It was concluded by all stakeholders at this meeting that, in light of the publication of the new requirements, any discussion of potential changes to the security cornerstone PI would be better informed after completion of one complete cycle of the baseline inspection program. Therefore, the staff plans to reassess the effectiveness of the security PI in 2013 as informed by the experience gained during the completion of one full security baseline inspection cycle.

The staff has also continued its efforts to improve the Emergency Preparedness PIs, specifically the Drill and Exercise Performance (DEP) PI. Data collection for Temporary Instruction (TI) 2515/175, "Emergency Response Organization, Drill/Exercise Performance Indicator, Program Review," issued June 5, 2008, has been completed. The staff's review of the collected raw DEP PI data is ongoing.

In addition to reinforcing the current PI guidance and improving existing PIs, the staff has made progress in exploring potential new PIs. The staff gained experience in the use of PIs outside the United States by participating as a consultant in a meeting of international regulators that completed a draft International Atomic Energy Agency safety guide titled "Development of Nuclear Power Plant Safety Performance Indicators for Use by a Regulatory Body." The staff also evaluated whether PIs already in use by the industry (i.e., non-ROP PIs) would provide meaningful regulatory insights that could be included in the ROP. A group of senior NRC inspection program managers reviewed the corporate and plant-specific indicators used by a licensee. The staff found that this licensee uses a large number of internal PIs, many of which involve information associated with NRC regulatory functions and activities. One area of interest was the use of MSPI insights to evaluate and plan potential plant modifications that, if implemented, would improve individual plant-risk profiles. Other than the MSPI, which the ROP already uses, the staff concluded that the other internal PIs either involved information and data that were too subjective for effective use as a regulatory tool or were not directly linked to regulated activities.

In December 2009, the staff introduced a white paper at the monthly public meeting of the industry-staff ROP Working Group that captured a broad spectrum of plant performance attributes, including those of PIs used by the international community, the United States nuclear power industry, and other organizations. The staff plans to host a separate public meeting to discuss the white paper in detail, with the goal of obtaining stakeholder input to develop a framework and establish a process for evaluating the efficacy of potential new PIs for use in the ROP.

Based on Commission direction in the staff requirements memorandum (SRM) dated June 30, 2008, the staff reviewed the metrics for assessing the effectiveness of the PI program and made revisions in CY 2008. The staff revised the wording to two metric definitions, as well as to the internal and external survey questions associated with them, to emphasize that the PI program is used in conjunction with the inspection program to provide useful insights (PI-4) and that the PI program contributes to the identification of performance outliers (PI-8). The results of the 2008 internal survey and recent 2009 external stakeholder survey indicate that the revisions to PI-4 and PI-8 helped to emphasize the role of the PI program, and the more accurate metric definition should ensure objective, open, and predictable future survey results. The staff is satisfied with the changes made to the PI metrics as a result of the CY 2008 review and considers the action complete to address the Commission SRM. In addition, the staff will continue to reinforce the message, through ongoing communications with both internal and external stakeholders, that the PI program is only a contributor to the identification of performance outliers and is used in conjunction with the inspection program to provide useful insights on licensee performance.

The staff met all eight of the PI metrics for CY 2009. This year, only industry stakeholders participated in the external survey though the survey was made available to all external stakeholders. This survey generally found that the PI program met the ROP goals of providing useful information on risk-significant areas. Most survey respondents found the PIs to be clearly

defined and understandable and to provide an appropriate overlap with the inspection program. In addition, the majority of the respondents stated that the PIs provide an objective indication of declining safety performance and contribute to the identification of outliers. Several respondents asserted that the MSPI is too complex, labor intensive, and difficult to understand. The NRC will endeavor to minimize the complexity of the MSPI when considering any future MSPI improvements. The staff will consolidate all responses to the external survey feedback in a separate document.

Inspection Program

The inspection program independently verified that licensees operated plants safely and securely in CY 2009 and identified and corrected performance issues in a timely manner in accordance with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program—Operations Phase," and IMC 2201, "Security and Safeguards Inspection Program for Commercial Nuclear Power Reactors." Each region documented its CY 2009 completion of the baseline inspection program in a memorandum available in ADAMS (Accession No. ML100390084 for Region I, ML100550802 for Region II, ML100560313 for Region III, and ML100601032 for Region IV). Additionally, the Office of Nuclear Security and Incident Response (NSIR) completed all security baseline inspections in CY 2009.

The staff completed its third biennial ROP realignment review during CY 2009, in accordance with Appendix B to IMC 0307, "Reactor Oversight Process Self-Assessment Program." This review assesses the effectiveness of each ROP baseline inspection procedure (IP) by determining whether appropriate inspection resources were applied in each of the inspectable areas. The working group consisted of staff from the Office of Nuclear Reactor Regulation (NRR), NSIR, and each of the four Regions. Modifications and adjustments to the inspection effort were made across the baseline inspection program, but overall inspection resources for CY 2010 remain at CY 2009 levels. The 2009 ROP realignment also added new inspection requirements to accommodate inspections related to the new requirements under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, "Fitness-for-Duty Programs," 10 CFR 50.54(hh), "Conditions of Licenses," and lessons learned from Peach Bottom regarding inattentive security officers. Additionally, the staff adjusted some IPs in the reactor safety area to better align budgeted and expended inspection resources. The staff revised all radiation safety inspection procedures to provide a more performance-based inspection for each of the functional areas of a radiation safety program. It also made inspection resource adjustments to all security-related IPs, based on regional feedback and past inspection resources expended for each IP. Additional details on the results of the 2009 ROP realignment process appear under ADAMS Accession No. ML092090312.

In addition, the NRC revised several inspection program documents and created one new IP to address Subpart I, "Managing Fatigue," of the new requirements contained in 10 CFR Part 26. For example, the staff added guidance to Appendix D, "Plant Status," to IMC 2515 for inspectors to look for indications of fatigue when performing plant status reviews, and created IP 93002, "Managing Fatigue," to provide guidance to inspectors for fatigue-related issues. In addition, the staff added an inspection requirement to IP 71111.20, "Refueling and Other Outage Activities," to determine how licensees manage fatigue during outages. The staff also revised the Security Baseline Inspection Program to address the new requirements of 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage," and is developing a new inspection program to address the new requirements of 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks."

The staff is developing a new engineering inspection to potentially replace the current component design-bases inspection. The new inspection will focus on improved component sample selection by reviewing various licensee programs and using operating experience (OpE). The staff plans an initial inspection during CY 2010, with full implementation of the new engineering inspection in CY 2011 if it is determined to be effective.

The inspection staff continued to review and consider OpE in planning their inspection activities. The Operating Experience Smart Sample (OpESS) Program provides inspectors with concise information related to selected industry operating events that have generic applicability and potential risk significance and can be readily inspected through the baseline inspection program. Since the program's inception in fiscal year (FY) 2007, the staff has issued seven OpESS documents and one update. The staff compiles and communicates operating experience using single Web page summaries called OpE COMMS, daily OpE Screening Summaries, OpE summary inputs and discussions during the monthly ROP call with the Regions, and the quarterly inspector newsletters. In addition, the Reactor OpE Gateway contains a wealth of OpE information for all internal stakeholders. This internal Web page includes numerous OpE data bases and search engines for all agency employees to use. The staff incorporated OpE into the assessment process for use during the mid-cycle and end-of-cycle reviews, as noted in the assessment program discussion below. The staff continues to emphasize the use of OpE and plans to further integrate this emphasis into the inspection program in CY 2010, through the development of a new IMC or incorporation into existing IMC guidance.

Although the resident inspector (RI) and senior resident inspector (SRI) turnover rates have declined for three consecutive years, the staff continues to closely monitor the attraction and retention of RIs and SRIs to ensure an experienced and stable RI and SRI program. An NRC senior-level management working group developed strategies and initiatives to address these retention issues and reported them to the Commission in SECY-09-0050, "Actions to Enhance Relocation and Retention for Employees," dated March 30, 2009. Enclosure 4 of this SECY paper offers additional discussion and analysis of resident inspector demographics and issues.

The staff continued to improve the initial and continuing inspector training programs to develop and maintain well-qualified, competent inspectors. The staff made recommendations, reviewed them in accordance with the ROP feedback process, and incorporated the improvements into inspection standards, as appropriate. The staff also developed three new inspector qualification standards, one for fire protection inspectors and two advanced-level standards for inservice inspection and fire protection inspectors. The staff conducted regional training on the integration of traditional enforcement into the assessment process, documenting issues in inspection reports, and licensee reporting requirements associated with the SSFF PI. In addition, the staff initiated periodic knowledge management seminars to improve the NRC's understanding of the concept of safety culture and its aspects. The staff also developed and implemented industrial safety training as well as a comprehensive training curriculum to support security inspections, including Force-on-Force inspections.

All inspection program metrics met their established criteria during CY 2009, including all timeliness goals. In general, respondents to the external survey believed the inspection program was effective in ensuring areas important to safety are appropriately addressed and that the information contained in inspection reports is relevant, useful, and clearly written. The agency received some feedback on potential areas for improvement which are addressed in this self-assessment and will be further addressed in the consolidated response to the CY 2009 external survey.

Significance Determination Process

The SDP continues to be an effective tool for determining the safety significance of identified performance issues. Oversight focuses on process improvements, based on feedback from internal and external stakeholders. The staff met the SDP timeliness metric for a fourth consecutive year. The staff received only one appeal letter, which was rejected because it failed to meet the criteria for invoking the appeal process. The goals met by the staff for other metrics included the amount of expended resources applied to SDP evaluations, compared to direct inspection hours, and ensuring that the SDP results are repeatable and predictable.

In CY 2009, the staff issued the new SDP for alternative mitigation strategies (Appendix L, "B.5.b Significance Determination Process," to IMC 0609, "Significance Determination Process.") The staff developed the appendix to support its commitment to the Commission to incorporate the lessons learned from the performance of Temporary Instruction 2515/171, "Verification of Site Specific Implementation of B.5.b Phase 2 & 3 Mitigating Strategies," into the ROP baseline inspection program. In a memorandum to the Commission, dated April 30, 2009 (ADAMS Accession No. ML090771056), the staff documented its fulfillment of this commitment. The staff revised and issued the baseline security SDP, Appendix E, Part 1, to enhance the process. The staff also updated the baseline security SDP to reflect the new requirements of 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage," and is working with stakeholders to revise the Force-on-Force (FOF) SDP to improve its effectiveness. In accordance with SRM M100112, "Briefing On Office Of Nuclear Security and Incident Response-Programs, Performance, And Future Plans", dated February 12, 2010, the staff will evaluate and report back to the Commission on how the proposed enhancements to the FOF physical protection SDP would alter the CY 2009 FOF exercise findings.

A team composed of staff members from NRR, the Office of Nuclear Regulatory Research (RES), and the Regions reviewed the NRC risk tools to identify areas for enhancement; this effort was called the partnering initiative. The team solicited feedback from internal stakeholders and end-users in the regional offices and Headquarters for improving the NRC risk tools used in everyday regulatory activities for nuclear reactors, such as the SDP, standardized plant analysis risk (SPAR) models, and the Incident Investigation Program, as well as staff training needs and interests. The team intends to use the data collected to (1) ensure the suite of risk tools is used efficiently, (2) provide clarity through improving documentation, methods, and training, (3) use the best available knowledge from research and operational experience to improve the suite of risk tools and thus improve the reliability and predictability of the NRC's performance assessment activities, and (4) provide better tools for all NRC staff engaged in probabilistic risk assessment (PRA) regulatory activities. The staff is currently working to implement many of these enhancements.

In the CY 2008 self-assessment, the staff agreed to develop and implement additional SDP training to ensure the inspectors remain efficient and effective in determining the safety and security significance of identified performance issues. Although the staff began to develop additional SDP training, it deferred implementation to incorporate input from the partnering initiative, which provided valuable insights regarding areas where training was lacking or can be improved. These areas include fundamental and overview training for certifying inspectors, as well as risk-informed decision making fundamentals and techniques for managers. The staff will resume its efforts to implement SDP training in CY 2010.

The staff continues to develop analytical tools that complement the NRC's deterministic approach and support its traditional defense-in-depth philosophy. Work on developing low power/shutdown (LPSD) SPAR models continues with a commitment of two models per year. Four LPSD models are currently available, with two more being developed and another one planned. Guidance for using the models appears in Volume 4 of the Risk Assessment of Operational Events (RASP) Handbook, which will be issued in CY 2010 for trial use and comment. Enclosure 2 of SECY-09-143, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated September 29, 2009, provides the status of these and other SPAR model enhancements.

Based on the Commission's direction in the SRM dated June 30, 2008, the staff reviewed the SDP metrics and made several changes, including the wording for the metric and the corresponding survey question regarding stakeholder perception (SDP-4), to emphasize that the SDP should result in an "appropriate" regulatory response across all cornerstones. The responses to this question in the external survey indicated that the stakeholders thought the SDP generally resulted in the appropriate regulatory response. The staff believes the changes made to the SDP metrics as a result of the CY 2008 review clarified the metric definition and intent of the SDP, and considers the action complete to address the Commission SRM.

Although the external survey responses were generally favorable, some stakeholders indicated that the SDP was not consistent and sometimes resulted in attributing higher risk significance to an issue than was warranted, that too much time was spent challenging NRC assumptions, and that the NRC should use licensee PRA models. The staff is considering these comments but fails to see evidence that the NRC's SDP results overestimate risk significance or that NRC assumptions are subjective having received only one appeal letter for findings of greater than Green significance, which was rejected because it failed to meet the criteria for invoking the appeal process. The staff will further address the survey responses in its consolidated response to stakeholder comments. The staff will continue to streamline the SDP program, implement effective staff training, and monitor SDP timeliness.

Assessment Program

Staff implementation of the assessment program ensured that staff and licensees took necessary actions to address performance issues in CY 2009. The staff revised IMC 0305, "Operating Reactor Assessment Program," to improve usability and incorporate added guidance on traditional enforcement and safety culture, as well as other clarifications and enhancements. In addition, to address its commitment in the 2009 ROP self-assessment, the staff incorporated into IMC 0305 consideration of operating experience during mid-cycle and end-of-cycle reviews to note trends in performance or the emergence of technical issues that can be considered for incorporation into ROP inspection guidance.

In addition to the changes to the IMC 0305 guidance, the staff enhanced the internal and external communication of plant assessment results. Part of this effort included a revision to the action matrix public Web site to support program changes included in the December 24, 2009, revision. Starting in CY 2010, the action matrix Web site will provide a more current status of plant assessment, rather than a purely retrospective look at the previous quarter's data. This change promotes clarity and openness with members of the public.

In its SRM M090514, "Briefing on the Results of the Agency Action Review Meeting," dated June 1, 2009, the Commission asked the staff to provide the status of the two facilities

(Davis-Besse and Indian Point) that were currently receiving increased NRC oversight as a result of deviations from the action matrix. In SECY-09-0121, "Status of the Deviation from the Reactor Oversight Process Action Matrix for Davis-Besse Nuclear Power Station and Indian Point Energy Center," dated August 24, 2009, the staff provided the plans and schedules for satisfying the criteria for these plants to return to normal NRC monitoring efforts. As of the end of CY 2009, the staff had closed out both the Davis-Besse and Indian Point deviations and there are some deviations from the action matrix in process in late CY 2009 and CY 2010.

As noted in the CY 2008 ROP self-assessment, the number of plants in the degraded cornerstone (column 3) and multiple/repetitive degraded cornerstone (column 4) was consistent with previous levels, and the industry's safety performance, as evidenced by the ROP, was consistent with the Industry Trends Program results. During CY 2009, the staff observed a decline in the number of plants in columns 3 and 4 of the action matrix. The staff will continue to closely monitor plant performance to ensure appropriate oversight.

In the CY 2008 ROP self-assessment, the staff committed to revising program documents to incorporate guidance for integrating traditional enforcement outcomes into the assessment process. During CY 2009, the staff completed efforts to integrate certain traditional enforcement items into the assessment program by changing inspection and assessment guidance documents. The staff changed Appendix B, "Issue Screening," to IMC 0612, "Power Reactor Inspection Reports," to allow performance deficiencies to be processed separately from the violation, so that the technical aspect can become a timely input into the action matrix. IMC 0305 and supporting inspection guidance were changed to allow follow up inspection on all levels of traditional enforcement outcomes. Using an escalating approach similar to that in the action matrix, the number, severity level, and similarities among the violations will allow one of three levels of inspection response to be used, as appropriate.

The staff committed, in the CY 2008 ROP self-assessment, to explore ways to use cross-regional experience to further improve the implementation of guidance on substantive cross-cutting issues (SCCIs). In response to this commitment, the staff leveraged ongoing efforts initiated by the Deputy Regional Administrators to improve the reliability of ROP implementation, including the SCCI process. Regional management developed the following four ROP reliability initiatives: (1) Enhanced Inspection Resource Sharing Among Regions, (2) Branch Chief Benchmarking Visits to Other Regions, (3) Periodic Discussion of Reliability Topics, and (4) ROP Self-Assessments of Inspection Report Quality. The regions are continuing to implement these initiatives, with NRR support.

The staff also committed, in the CY 2008 ROP self-assessment, to revising program guidance, as necessary, to better align with the Commission's safety culture policy statement once it has been completed. While the Commission safety culture policy statement is being developed, staff continues to be engaged with internal and external stakeholders. In addition, the Nuclear Energy Institute (NEI) has proposed an alternative industry-owned safety culture oversight process, aspects of which the NRC staff is observing at the request of NEI. The staff will continue to become familiar with the initiative and to evaluate associated tools that could possibly be leveraged to gain efficiencies in the ROP.

The staff implemented several changes to ROP guidance in CY 2009 regarding safety culture and the use of SCCIs. It revised IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," in January 2009. This revision incorporated a graded approach for assessing a licensee's safety culture and detailed guidance for performing an independent safety culture assessment. Based

on regional experience and feedback, the staff revised the IMC 0305 guidance to create a cross cutting theme for the area of safety-conscious work environment (SCWE). In addition, to improve program document usability, guidance related to screening inspection findings for cross-cutting aspects was relocated from IMC 0305 to IMC 0612, Appendix B, and the descriptions of the safety culture components and aspects were relocated to the new IMC 0310, "Components Within the Cross-Cutting Areas." While guidance for screening inspection findings and the component descriptions were relocated, IMC 0305 retained all guidance related to the SCCI process. The staff also developed training for regional staff on the NRC's safety culture activities related to the ROP.

The staff believes that the current process of considering cross-cutting aspects of inspection findings is effective because it offers insights into a licensee's safety culture, while maintaining consistency with the ROP objectives of being transparent, objective, understandable, predictable, risk-informed, and performance-based. The process enables the NRC staff to identify concerns about a licensee's performance in a cross-cutting area, with the expectation that the licensee will address the performance issue before it results in a more significant safety concern.

The agency met seven of the eight assessment metrics for CY 2009, including all timeliness goals. The metric regarding perceived effectiveness of the safety culture enhancements to the ROP was not met, based on the negative feedback from external stakeholders, which included only five responses, all from industry representatives. The staff is aware of the industry's concerns with the process for determining substantive cross-cutting issues and will continue to consider industry proposals as noted above. The staff also recognizes that there was a significant decrease in the number of external survey responses and notes that it would be prudent to obtain a broader perspective before drawing specific conclusions on the process. Other feedback from the external survey regarding the assessment program was generally favorable. Respondents confirmed that actions taken to address performance issues at plants are predictable and appropriate, and that information contained in assessment reports is, for the most part, relevant, useful, and well written. Some respondents questioned whether multiple White inputs should move a plant to column 3 and encouraged greater consistency and clarity on substantive cross-cutting issues. The staff will respond to specific comments as part of its consolidated response to the external survey.

Regulatory Impact Summary

Scope and Objectives

On December 20, 1991, the Commission issued a staff requirements memorandum directing the staff of the U.S. Nuclear Regulatory Commission (NRC) to develop a process for obtaining continual feedback from licensees and to report the feedback on the process to the Commission each year. The staff described the continual feedback process in SECY-92-286, "Staff's Progress on Implementing Activities Described in SECY-91-172, 'Regulatory Impact Survey Report—Final,'" issued August 18, 1992.

The feedback process requires regional management to solicit informal feedback from its licensees during routine visits to reactor sites. The managers record this feedback on forms that they forward to the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Security and Incident Response (NSIR). The Regions, NRR, and NSIR then evaluate the concerns identified and take any necessary corrective actions. This process, first implemented in October 1992, has given licensees frequent opportunities to comment on the NRC's regulatory impact.

This enclosure reports on feedback received from licensees during the previous fiscal year. During this period, the staff received and compiled feedback from 95 site visits to 43 reactor sites (68 units) across all four regions. These visits resulted in 178 distinct comments that fell into two main categories—formal communications with licensees and inspector performance. Of the comments compiled, 92 percent (163/178) were favorable and 8 percent (15/178) were unfavorable. The number and distribution of comments and the favorable percentage were similar to previous years. The following sections summarize the feedback received, the staff's evaluation, and the proposed improvement actions.

(1) Formal Communications with Licensees

Feedback

Almost half of the licensees' comments concerned the effectiveness of communications between the NRC staff and licensees. Almost all comments were favorable with regard to communications with inspectors and regional management. Many licensees said that communications were good or excellent, and others noted that the staff's communication skills have improved. A few licensees noted communication concerns with inspection staff, one noted that the NRC did not respond well to questions during a public meeting with the licensee, and another licensee noted that the NRC's preliminary determination letter provided before an enforcement conference was not properly characterized.

Evaluation and Action

The staff concludes that communication between the NRC and its licensees is effective and that the reported communication problems were isolated instances. The staff bases this conclusion on the large number of routine interactions between the NRC and its licensees, combined with the many favorable comments and the relatively few negative comments received during the past year. Nearly 95 percent of the comments received this year were favorable.

The staff is aware of the importance of prompt and clear communication and emphasizes this goal in the policy, guidance, and training provided for the inspection program. Effective

communications will remain a priority and will receive continued monitoring and attention from regional and headquarters management.

(2) Inspector Performance

Feedback

Almost half of the licensees' comments concerned inspector performance. This category covers a wide range of inspector practices but excludes issues involving communication with licensees discussed in the previous section. Nearly all of the comments were complimentary of the NRC's inspection staff, noting the high quality of inspections and the effective and professional working relationship between the NRC and its licensees. Most licensees noted that NRC inspections were effective, and the associated inspection reports correctly characterized the licensee's performance. However, a few licensees questioned the NRC's basis for specific violations related to three distinct inspections (fire protection; modifications under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59, "Changes, Tests, and Experiments"; and keeping exposures as low as reasonably achievable), and another licensee questioned the staff's timeliness in dispositioning a piping issue.

Evaluation and Action

The staff concludes that inspectors were professional, maintained effective working relationships, and appropriately characterized licensee performance. Over 95 percent of the comments received this year were favorable. The staff reviewed the negative feedback for trends and found that each concern related to an isolated incident or a difference in professional opinion. As stipulated in Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)," to Inspection Manual Chapter 0609, "Significance Determination Process," the significance determination process has a formal venue for a licensee to appeal the staff's final significance determination of an inspection finding. This process was invoked only once in 2009, and that appeal letter was rejected because it failed to meet the criteria for invoking the appeal process.

The NRC management continues to emphasize to the staff the importance of professional conduct. Senior NRC managers reinforce these expectations in inspector counterpart meetings, workshops, and training courses, as well as during site visits conducted in accordance with Inspection Manual Chapter 0102, "Oversight and Objectivity of Inspectors and Examiners at Reactor Facilities." The staff will continue to closely monitor the regulatory impact of inspector performance.

Reactor Oversight Process Resources

Summary of 2009 Reactor Oversight Process Resources

Table 1 summarizes the U.S. Nuclear Regulatory Commission (NRC) staff resources expended, in hours, for the Reactor Oversight Process (ROP) during the past three fiscal years (FYs).[1] Overall staff effort in FY 2009 increased by 1.4 percent, compared with FY 2008, for the activities listed in Table 1.

Baseline inspection hours include direct inspection effort, baseline inspection preparation and documentation, and plant status activity. Baseline inspection hours increased in 2009, primarily as a result of increased effort in performing Inspection Procedure (IP) 71152, "Identification and Resolution of Problems," and IP 71130.03, "Contingency Response—Force-on-Force Testing." The hours charged to other baseline procedures remained relatively unchanged. As in previous years, all four Regions completed the required baseline inspections in 2009.

Plant-specific inspections include: (1) supplemental inspections conducted in response to greater-than-Green inspection findings and performance indicators, (2) reactive inspections, such as augmented inspection teams and special inspections performed in response to events, and (3) the infrequently performed inspections listed in Appendix C, "Special and Infrequently Performed Inspections," to NRC Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program—Operations Phase," and Appendix C, "Generic, Special, and Infrequent Inspections," to IMC 2201, "Security Inspection Program for Commercial Nuclear Power Reactors," which are not part of the baseline or supplemental inspection programs.

Plant-specific inspection effort decreased in FY 2009, compared with FY 2008, caused, in part, by completion of IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," at the Palo Verde site. There was also a decrease in event response and in supplemental inspections in FY 2009, with a corresponding decrease in preparation and documentation effort for these inspections. The decrease in supplemental inspections reflects the decrease in the number of plants in columns 3 and 4 of the ROP action matrix in FY 2009, compared with FY 2008.

Generic safety issue inspections are typically one-time inspections of specific safety and security issues, with significant variability in effort possible from year to year. The decreased effort in generic safety issue inspections reflects reduced activity in this area in FY 2009 and the completion, in December 2008, of inspections related to "Verification of Site Specific Implementation of B.5.b Phase 2 & 3 Mitigating Strategies."

The effort reported for "other activities," including inspection-related travel, significance determination process (SDP), and routine communications (which now encompasses regional support, enforcement support, and review of technical documents), increased slightly in 2009. The effort for these activities typically corresponds to the baseline inspection effort. The regional effort for licensee performance assessments has remained relatively steady during the past three FYs and suggests that the performance assessment effort has reached a steady state.

[1] The staff implements the ROP on a calendar year basis; however, it obtains and reports resource data on an FY basis.

ROP Resource Improvement Initiatives

The staff continued to implement a number of initiatives to improve program efficiency and effectiveness to reduce inspection resource requirements. These initiatives include a realignment of resources allocated to the individual baseline IPs (including design engineering inspections), regional best practice initiatives (including the ROP reliability initiatives), continued SDP improvements, and implementation of the performance indicator improvements. Enclosure 1 of this paper discusses these initiatives.

Table 1 Resources Expended[1]
(Inspection-Related Staff Effort Expended at Operating Power Reactors)

	52 weeks FY 2007 09/24/06–09/22/07	52 weeks FY 2008 09/23/07–09/20/08	52 weeks FY 2009 09/21/08–09/19/09
Baseline Inspections			
Direct Inspection Effort	156,547	147,396	156,348
Inspection Prep/Doc	111,770	99,522	104,825
Plant Status	48,804	49,481	50,192
Subtotal	317,121 hr	296,399 hr	311,365 hr
Plant-Specific Inspections			
Direct Inspection Effort	12,278	14,063	9,149
Inspection Prep/Doc	8,174	9,909	6,338
Subtotal	20,452 hr	23,972 hr	15,487 hr
Generic Safety Issues Inspections	11,212 hr	13,492 hr	8,619 hr
Performance Assessment (Regional Effort Only)	14,349 hr	13,517 hr	15,478 hr
Other Activities[2]	68,493 hr	65,754 hr	67,972 hr
Total Staff Effort	431,627 hr	413,134 hr	418,921 hr
Total Staff Effort/Operating Site	6,540 hr/site	6,260 hr/site	6,347 hr/site

1 Resources expended include regional, NRR, and NSIR hours.
2 Other activities consist of inspection-related travel, SDP, and routine communications (which include regional support, enforcement support, and review of technical documents).

Resident Inspector Demographics

Scope and Objectives

This enclosure is the annual update on demographic data for inspectors assigned to the resident inspector (RI) program, requested by the Commission in its staff requirements memorandum (SRM) for COMGJD-98-001/COMEXM-98-002, "Discussion of Resident Inspector Demographics and the Balance Between Expertise and Objectivity," issued April 8, 1998. This analysis seeks to determine whether the actions of the U.S. Nuclear Regulatory Commission (NRC) associated with the RI program have resulted in a stable or increasing RI experience base and to identify any necessary improvements. This enclosure also provides an update on site staffing.

Resident Inspector Demographic Data

The NRC's staff review of the demographics included an analysis of the overall program data for the RI and senior resident inspector (SRI) groups (see Tables 1–7 and Figures 1–10). Inspection Manual Chapter (IMC) 0307, "Reactor Oversight Process Self-Assessment Program," includes details regarding the RI program demographic data analysis. The staff used median values from the month of November in 2005 for each year through 2009 for statistical comparison.

The demographic analysis consists of the following four distinct data sets:

(1) "NRC time" is the total number of years the individual has accumulated as an NRC employee.

(2) "Total resident time" is the total number of years the individual has accumulated as an RI or SRI.

(3) "Current site time" is the total number of years spent as an RI or SRI at the current site.

(4) "Relevant non-NRC experience" is nuclear power experience acquired outside the NRC. Examples of relevant non-NRC experience include operation, engineering, maintenance, or construction experience with commercial nuclear power plants, naval shipyards, U.S. Department of Energy facilities, or the U.S. Navy's nuclear power program.

Analysis of the 2009 Resident Inspector Group

The RI demographic data for 2009 (see Table 1) indicates that the RI turnover rate has been on a downward trend from 2007 through 2009 (46 percent, 31 percent, and 22 percent). This is significant, given the 46-percent turnover rate in 2007, compared with 20 percent for 2006. Of the 15 RIs who left during 2009, 6 were promoted to SRI positions, 7 were either promoted or laterally reassigned to a regional office or Headquarters, and 2 resigned from the NRC.

The high turnover in 2007 resulted in about half of the RIs being in new assignments, which likely contributed to the reduced turnover in the following two years. In addition, the current real estate market has been a negative incentive for turnover, and caused several SRIs and RIs to apply for extensions beyond seven years. Finally, as discussed later in this enclosure, the staff has implemented a number of initiatives to attract and retain resident inspectors which may also

have contributed to the reduction in turnover. The staff will continue to monitor the affect of these initiatives on resident staff turnover.

Concurrent with the reduction in 2007 through 2009, NRC time (nationally) has steadily increased, and relevant non-NRC experience has steadily decreased (Table 2). Both of these trends may have resulted from the 2007–2009 turnover reduction. Table 6 shows a breakdown of experience data for RIs by region. This table shows that Region II has significantly greater relevant non-NRC experience than the other regions.

Table 1 RI Turnover

	2005	2006	2007	2008	2009
Promoted to SRI	10	11	13	10	6
Promoted/ Reassigned	9	2	13	8	7
Retired	2	1	3	1	0
Resigned	2	0	4	3	2
Total	**23**	**14**	**33**	**22**	**15**
Turnover Rate	32%	20%	46%	31%	22%

Table 2 RIs
(Median Values in Years)

	2005	2006	2007	2008	2009
NRC Time	3.36	4.04	4.25	4.48	5.42
Total Resident Time	2.31	2.39	1.87	1.28	1.79
Current Site Time	2.25	2.23	1.85	1.28	1.79
Relevant Non-NRC Experience	10.63	10.75	10.38	9.00	6.25

Analysis of the 2009 Senior Resident Inspector Group

SRI demographic data for 2009 (see Tables 3 and 4) indicate that the SRI turnover rate for 2007 through 2009 steadily declined (26 percent, 18 percent, and 11 percent). The factors that influenced the reduction in RI turnover discussed previously also likely influenced the reduction in SRI turnover. In 2009, 7 of 66 SRIs left their SRI position at a specific site. Of these, 4 were promoted, 2 were reassigned (including SRIs who were laterally reassigned to another site),

and 1 resigned from the NRC. Table 4 indicates little variation nationally for the experience criteria. However, Table 7 indicates wide variance among regions for all but current site time.

Table 3 SRI Turnover

	2005	2006	2007	2008	2009
Promoted	5	7	7	5	4
Reassigned	4	7	7	4	2
Retired	1	1	1	1	0
Resigned	0	1	2	2	1
Total	**10**	**16**	**17**	**12**	**7**
Turnover Rate	15%	24%	26%	18%	11%

Table 4 SRIs
(Median Values in Years)

	2005	2006	2007	2008	2009
NRC Time	8.84	9.28	10.11	10.86	10.06
Total Resident Time	7.54	7.77	7.93	6.78	7.71
Current Site Time	2.63	3.21	2.52	2.28	2.44
Relevant Non- NRC Experience	7.96	9.08	10.04	9.38	9.51

Resident Inspector Attraction and Retention

Staff turnover within the NRC, whether caused by promotion, reassignment, retirement, or resignation, is an ongoing process from which the RI program is not insulated. To ensure that the RI program can continue to fulfill its mission, the Commission directed the staff in SRM M070531, "Briefing on the Results of the Agency Action Review Meeting (AARM)," dated June 14, 2007, to evaluate recruitment, training, and development to confirm that there are adequate human resources to meet changing needs. Therefore, because of the importance of maintaining an experienced and stable onsite inspection presence, the NRC initiated several actions to help alleviate the burden associated with the transient nature of the RI program.

SECY-09-0050, "Actions to Enhance Relocation and Retention for Employees," dated March 30, 2009, informed the Commission of staff actions to enhance the relocation and

retention of employees. The staff identified existing authorities and flexibilities that could be further developed and appropriately used to enhance the agency's current relocation and retention processes. Some of the enhancements, initially considered in connection with the RI program, may apply to other agency positions for which the agency might need to enhance its efforts to relocate or retain employees in the future.

In its SRM dated June 26, 2009, the Commission reaffirmed the 7-year rotation policy for SRIs and RIs and approved the staff's proposals to use existing authorities to enhance the agency's current relocation and retention processes to address the turnover in SRI and RI positions. The SRM asked the staff to report to the Commission within 2 years on the effectiveness of these changes.

Site Staffing

The staff developed a site staffing metric of 90 percent programwide, in response to a recommendation by the Davis-Besse Lessons Learned Task Force (DBLLTF). The purpose of the metric is to evaluate the agency's ability to provide continuity of regulatory oversight through the timely assignment of permanent RI staff. Specifically, DBLLTF Item 3.3.5.3 recommended that the staff establish a measurement for RI staffing, including program expectations, to satisfy minimum staffing levels. IMC 0307 provides details regarding the site staffing metric and criterion.

Despite the turnover rates in the RI and SRI positions, the regions succeeded in meeting their site staffing metric of 90 percent. The average site staffing for all regions was 97.55 percent in calendar year 2009. However, five sites fell below the 90-percent site staffing requirement, though these sites were not recurrences from the previous year. All five sites were staffed above 76 percent and were supplemented by region-based inspectors to assist in completing the baseline inspection program. Meeting this metric was challenging and had a significant impact on inspectors and management, but the recent relocation and retention enhancements may improve future site staffing metric results. Table 5 tracks the number of sites since 2005 that did not meet the 90-percent site staffing goal.

Table 5 Number of Sites Under 90-Percent Site Staffing

	2005	2006	2007	2008	2009
Number of Sites	3	1	9	5	5

Table 6 RIs 2009 by Region
(Median Values)

2009	NRC Time (years)	Total Resident Time (years)	Current Site Time (years)	Relevant Non-NRC Experience (years)
Region I	5.45	1.63	1.63	6.42
Region II	5.18	2.34	2.34	11.46
Region III	5.24	1.65	1.65	5.96
Region IV	5.27	1.79	1.79	6.00
All Regions	**5.42**	**1.79**	**1.79**	**6.25**

Table 7 SRIs 2009 by Region
(Median Values)

2009	NRC Time (years)	Total Resident Time (years)	Current Site Time (years)	Relevant Non-NRC Experience (years)
Region I	14.47	7.19	2.55	7.41
Region II	8.53	7.9	2.26	12.83
Region III	12.08	11.16	3.51	7.0
Region IV	7.42	5.31	2.28	9.42
All Regions	**10.06**	**7.71**	**2.44**	**9.51**

Conclusions

The staff has concluded that sites continue to be staffed with knowledgeable and experienced RIs and senior resident inspectors (SRIs). The demographic data indicate that:

- there is an improving trend in the turnover rate for both SRIs and RIs as indicated in Tables 1 and 3.
- regional training efforts ("inspector pipelines") are having a positive impact on the NRC experience level for RIs as indicated in Figure 1.

In addition, feedback from licensees noted that the inspectors performed high quality and effective inspections that correctly characterized the licensee's performance (as discussed in Enclosure 2, "Regulatory Impact Summary").

Many of the RI program incentives described in SECY-09-0050 have only recently been implemented or are in the process of being implemented. Therefore, improvements in the RI demographics are expected to continue. Notwithstanding, the NRC will continue to monitor SRI and RI staffing and retention to identify any adverse trends early.

The effectiveness of the enhancements to the relocation and retention initiatives described in SECY-09-0050 will be discussed in a separate paper to the Commission in CY 2011 in accordance with its associated SRM dated June 26, 2009.

(1) **NRC Time**: NRC time for the RIs increased about the same for all regions from 2008 to 2009, as indicated by parallel lines on the graph. NRC time for the SRIs increased in Regions I and III, decreased in Region II, and remained relatively constant in Region IV.

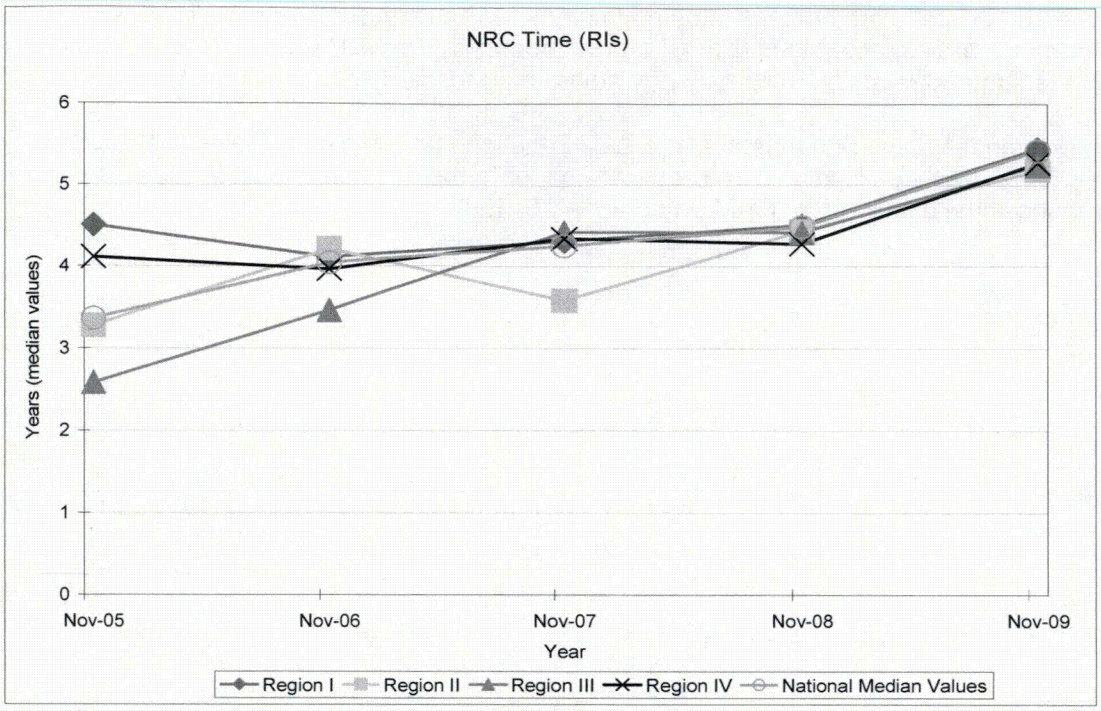

Figure 1

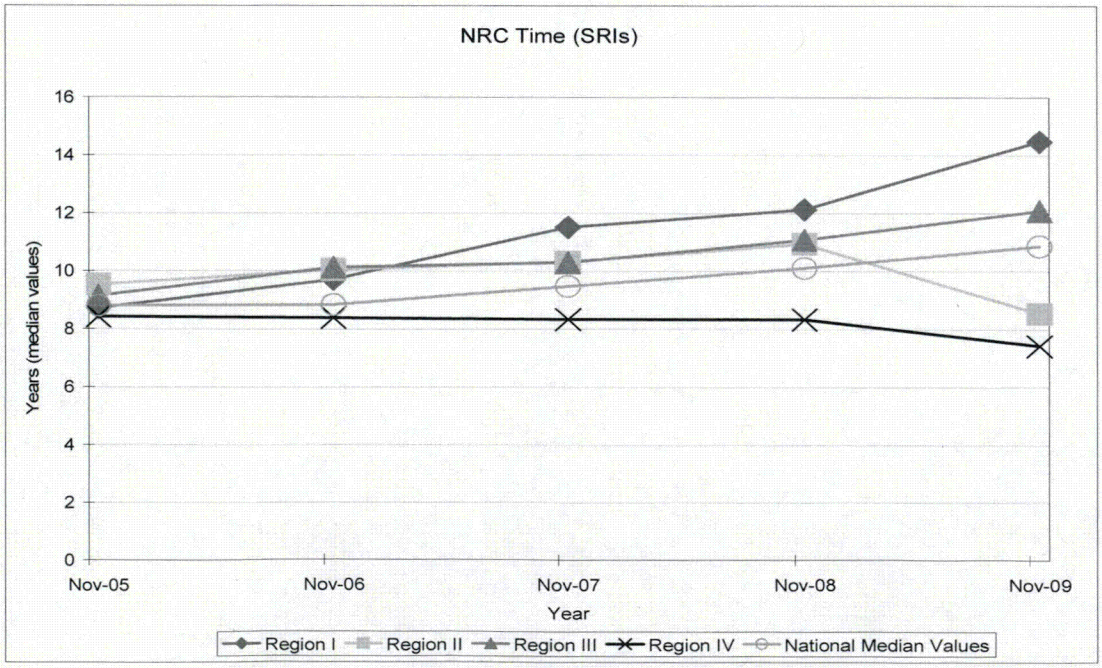

Figure 2

(2) **Total Resident Time**: From 2008 to 2009, total resident time for the RIs increased in all regions. Total resident time for the SRIs decreased in Region II and increased in the other regions.

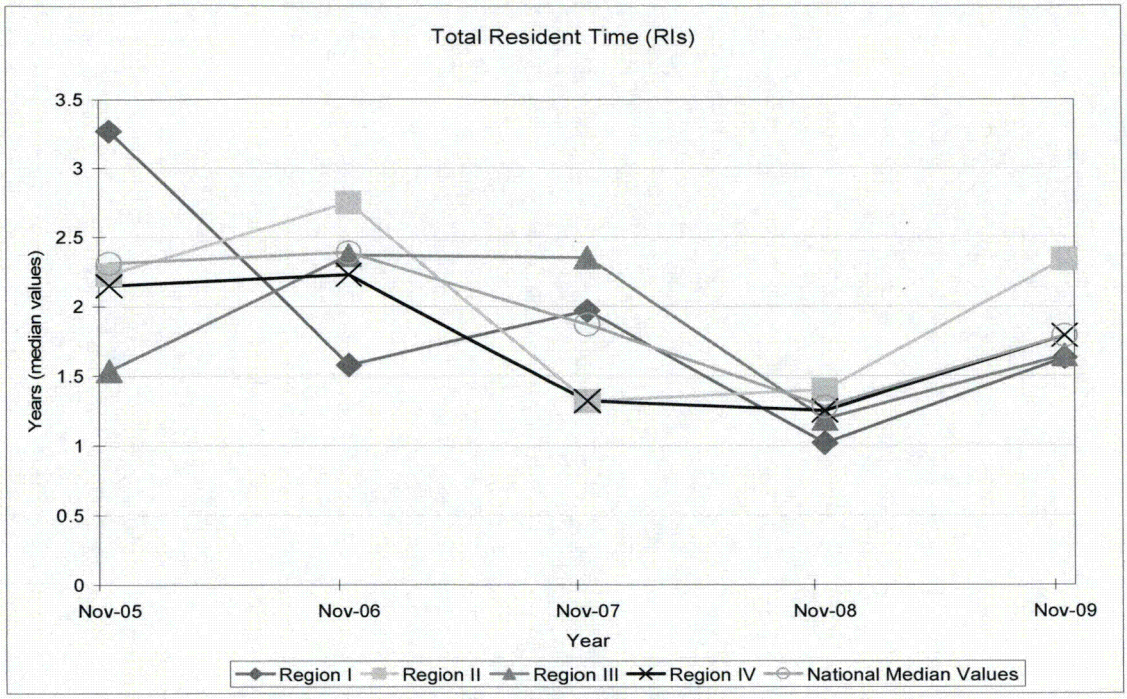

Figure 3

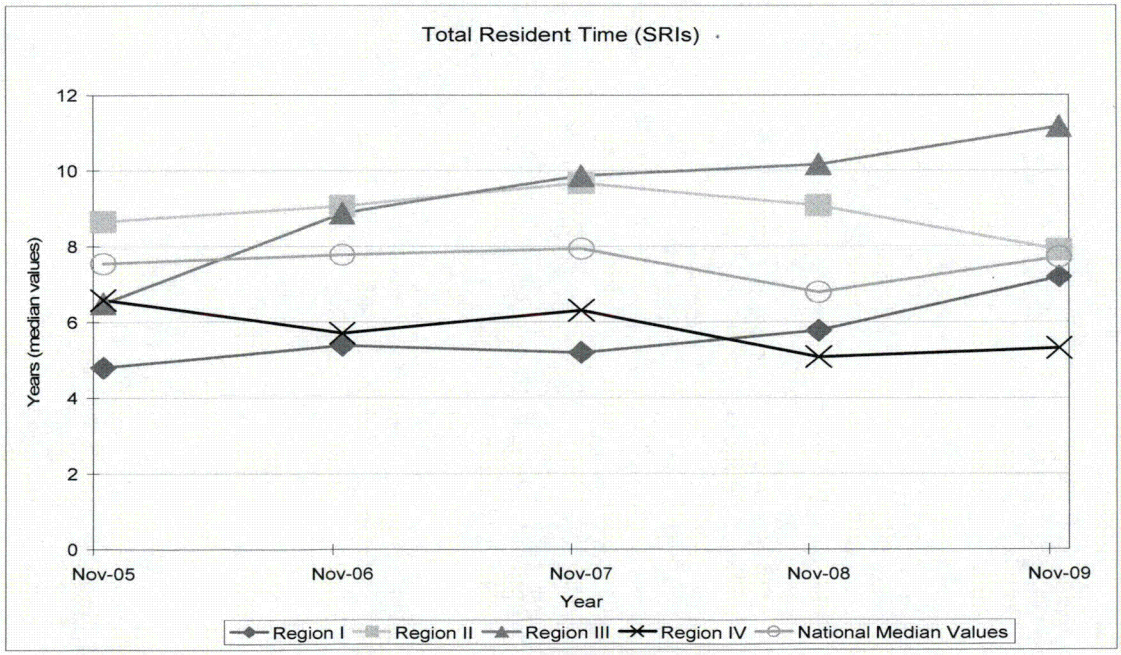

Figure 4

(3) **Current Site Time**: From 2008 to 2009, current site time for the RIs increased in all regions. Current site time for the SRIs increased in Regions I, III, and IV, and decreased in Region II.

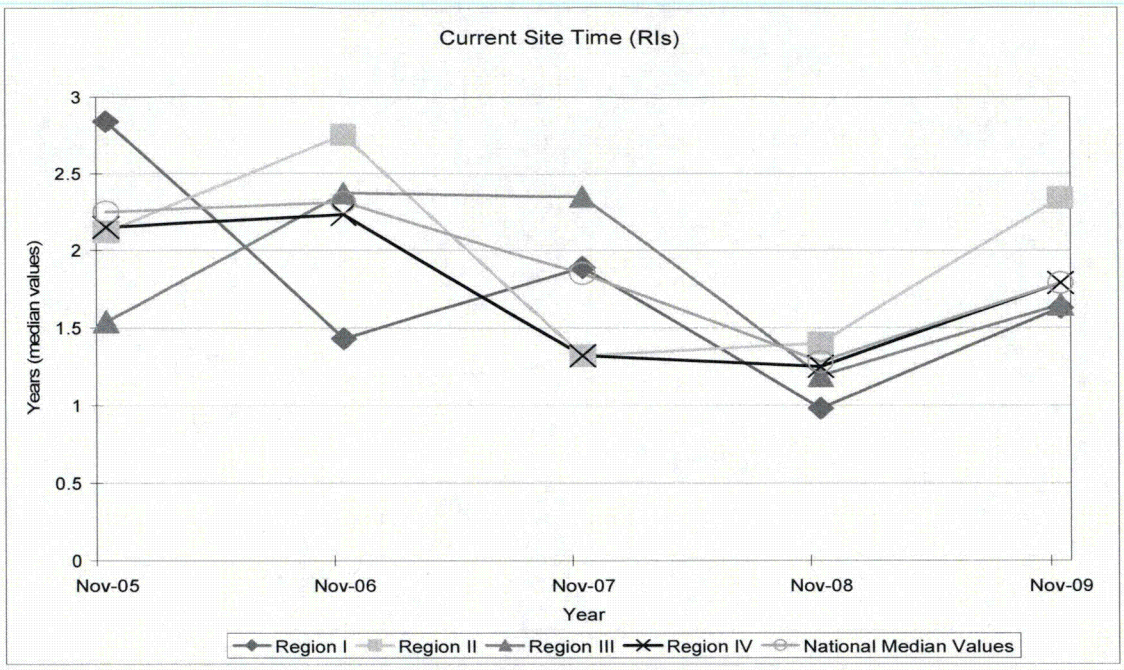

Figure 5

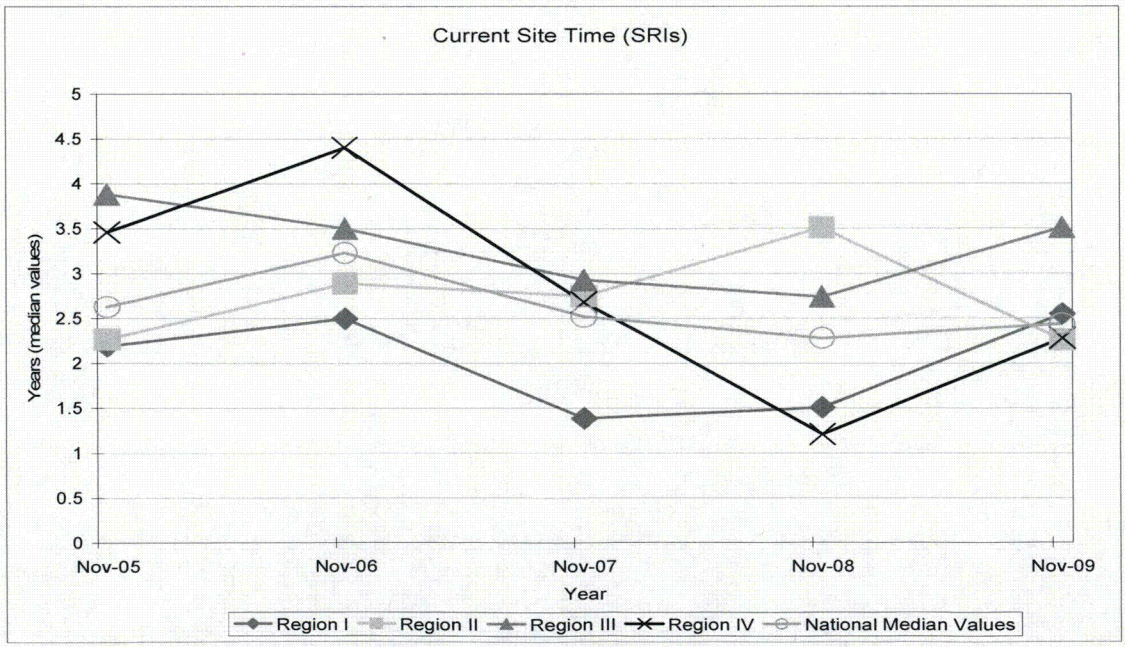

Figure 6

(4) **Relevant Non-NRC Experience:** From 2008 to 2009, relevant non-NRC experience for the RIs decreased in Regions I, II and III, and remained relatively constant in Region IV. Relevant non-NRC experience for the SRIs decreased in Region III and remained relatively constant in the other regions.

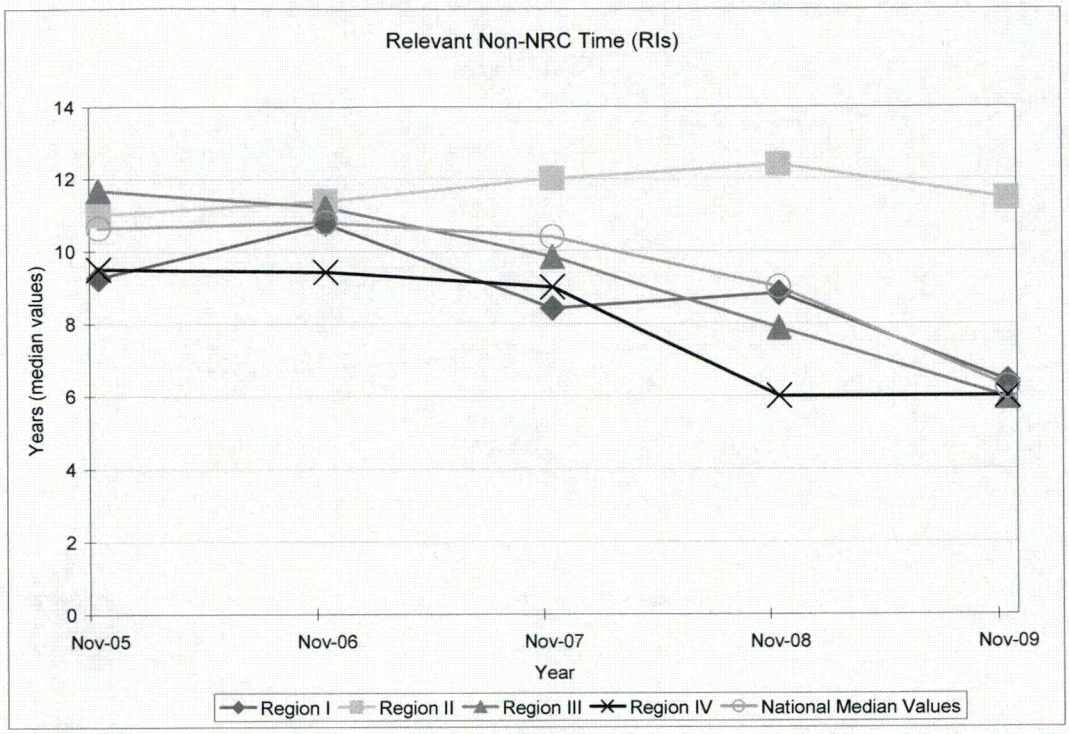

Figure 7

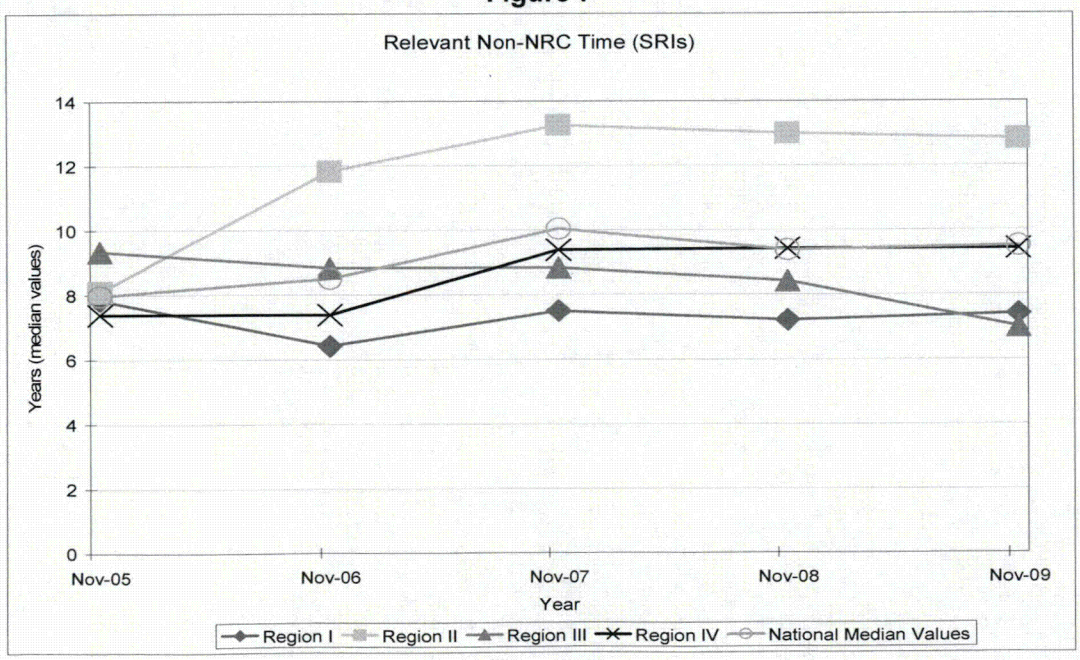

Figure 8

(5) **Summary:** Figures 9 and 10 graphically portray the average national demographic data for the RIs and SRIs shown in Tables 2 and 4.

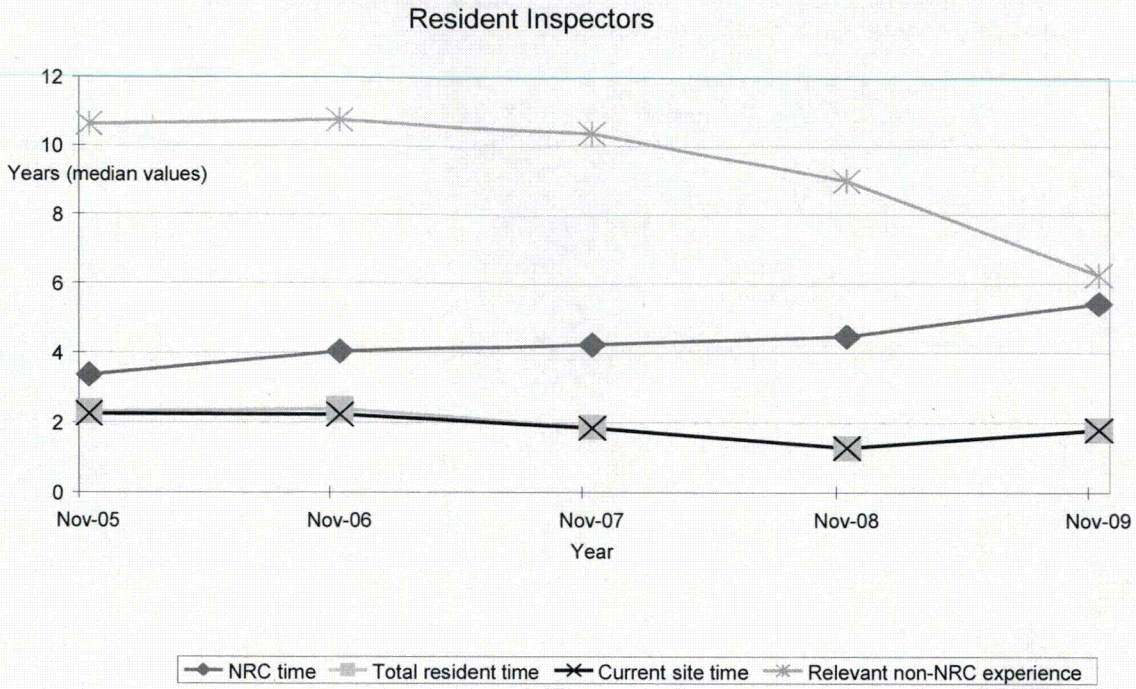

Figure 9

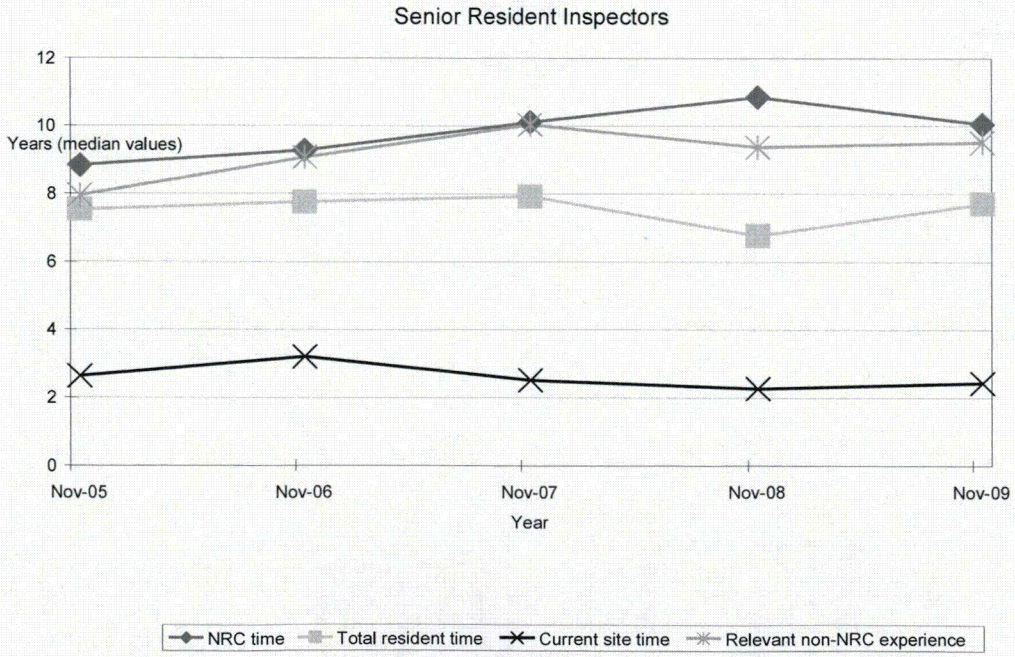

Figure 10

POLICY ISSUE
(Information)

August 5, 2010 SECY-10-0100

FOR: The Commissioners

FROM: Michael R. Johnson, Director
Office of New Reactors

SUBJECT: STAFF PROGRESS IN RESOLVING ISSUES ASSOCIATED WITH INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

PURPOSE:

The purpose of this paper is to inform the Commission of progress made by the staff in resolving issues associated with inspections, tests, analyses, and acceptance criteria (ITAAC), including the progress that has been made on developing a process to ensure that the validity of conclusions regarding acceptability of completed ITAAC is maintained.

This paper does not address any new commitments or resource implications.

SUMMARY:

The staff is making good progress on its efforts to develop a proposed rule and guidance to address ITAAC maintenance during the period between a licensee's submittal of an ITAAC completion letter and the Commission's Title 10 of the *Code of Federal Regulations,* Part 52, Section 103(g) (10 CFR 52.103(g)) finding. The staff plans to provide a proposed rule to the Commission in August 2010, and issue the associated draft guidance for public comment shortly after publication of the proposed rule. The staff also completed an evaluation of when changes to structures, systems, and components (SSCs) or emergency preparedness (EP)

CONTACT: James E. Gaslevic, NRO/DCIP
301-415-2776

Mahmoud M. Jardaneh, NRO/DCIP
301-415-2753

program elements related to closed ITAAC would necessitate license amendments and is developing guidance to provide additional clarity. The staff has been developing an ITAAC Closure Verification Process (ICVP) and associated implementing documents to ensure staff readiness for verification of closed ITAAC. The U.S. Nuclear Regulatory Commission (NRC) staff, the U.S Department of Energy (DOE), Southern Nuclear Company, and Westinghouse are planning to conduct a demonstration project to exercise the ICVP through a number of simulated ITAAC closure activities. The staff will use insights gained from the exercise to enhance the ICVP, as necessary. The staff is also updating Regulatory Issue Summary (RIS) 2008-05, "Lessons Learned to Improve Inspections, Tests, Analyses, and Acceptance Criteria Submittal," dated February 27, 2008, to include additional lessons learned from recent and ongoing reviews of applications. The Office of New Reactors (NRO) has developed and is providing training to the technical review staff on these lessons learned in this area and is communicating these insights to industry and stakeholders.

BACKGROUND:

In SECY-08-0117, "Staff Approach To Verify Closure of Inspections, Tests, Analyses, and Acceptance Criteria and To Implement Title 10 CFR 52.99, 'Inspection during Construction,' and Related Portion of 10 CFR 52.103(g) on the Commission Finding," dated August 7, 2008, (Agencywide Documents Access and Management System (ADAMS) Accession No.(ML081220237), the staff provided an update on plans to inspect and perform the closure verification of licensee-completed ITAAC. The staff also described how it intends to implement 10 CFR 52.99, "Inspection during construction," and the related provision of 10 CFR 52.103(g) regarding the Commission finding on whether all ITAAC acceptance criteria are met. The staff requirements memorandum related to SECY-08-0117, dated January 14, 2009, (ADAMS Accession No. ML090140136), directed the staff to keep the Commission informed of progress in resolving issues associated with ITAAC, including instances where successfully completed ITAAC are no longer satisfied.

In SECY-09-0119, "Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria," dated August 26, 2009, (ADAMS Accession No. ML091980327), the staff discussed progress toward resolving issues concerning ITAAC maintenance[1] and reporting, including the notification thresholds for events that may invalidate a previous determination that an ITAAC has been successfully completed. The staff also provided an update on its approach for making its recommendation to the Commission regarding the finding under 10 CFR 52.103(g) on whether all ITAAC in the combined license (COL) are met.

The staff hosted nine public workshops in the last 12 months to solicit input and exchange views on issues related to ITAAC closure and maintenance associated with previously successfully completed ITAAC. The agency held a Category II public meeting on July 29, 2010, to discuss the ITAAC closure and verification demonstration exercise. Members of the public, the Nuclear Energy Institute (NEI), industry representatives, and other external stakeholders participated in these public workshops.

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1 ITAAC Maintenance applies after ITAAC has been completed and provides confidence that the ITAAC continue to be met and that no activity has invalidated the basis for determining that the ITAAC are met.

DISCUSSION:

ITAAC Maintenance Regulatory Guidance and Rulemaking Progress

In SECY-09-0119, the staff informed the Commission of its progress in resolving issues concerning ITAAC maintenance and reporting, including the notification thresholds for events that may invalidate a previous determination that an ITAAC has been successfully completed. The staff, through public workshops, has made significant progress in developing and refining the notifications for completion of all ITAACs and notification of ITAAC maintenance issues.

The staff has prepared a proposed rule and will soon seek the Commission's approval of its proposal to amend the regulations in 10 CFR 52.99. Specifically, the staff will propose new provisions that apply after a licensee has completed an ITAAC and has submitted an ITAAC closure letter (ICL) to the NRC. The new provisions would require the licensee to (1) report new information that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met, (2) document the basis for all ITAAC notifications, and (3) notify the NRC of completion of all ITAAC activities. The staff plans to submit this proposed rule for Commission review in August 2010 and, if approved by the Commission, would issue the proposed rule for public comment following the incorporation of any Commission comments.

The staff is also preparing Revision 1 to Regulatory Guide 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52," issued October 2009, to incorporate guidance associated with ITAAC maintenance. The staff plans to issue the proposed guidance to address ITAAC maintenance and supplemental reporting for public comment concurrent with or shortly following the proposed rule publication.

License Amendments Necessitated by Changes to ITAAC

To address Commission comments from the staff's last briefing on ITAAC maintenance issues on September 22, 2009, the staff has evaluated when changes to SSCs or EP program elements related to closed ITAAC result in the ITAAC not being met such that a license amendment would be necessary. License amendments related to ITAAC performance are not unique to ITAAC maintenance. A license amendment would be necessary, even in the performance of the original ITAAC, if the licensee cannot perform the "prescribed" inspections, tests, or analyses or show that the acceptance criteria have been met.

The regulation at 10 CFR 52.98(f) states, "Any modification to, addition to, or deletion from the terms and conditions of a combined license, including any modification to, addition to, or deletion from the inspections, tests, analyses, or related acceptance criteria contained in the license is a proposed amendment to the license. There must be an opportunity for a hearing on the amendment." For amendment of an ITAAC originating from the referenced certified design, the licensee would also request an exemption from the standard design certification (DC) rule pursuant to 10 CFR 52.99(d)(1).

Enclosure 1 to SECY 09-0119 presented four reporting thresholds to identify whether activities would materially alter the ITAAC determination bases during the ITAAC maintenance period. The staff used these reporting thresholds as a basis for considering whether a license amendment would be necessary during the ITAAC maintenance period.

During a public workshop held on December 17, 2009, the staff presented and discussed with stakeholders events or activities associated with the ITAAC maintenance thresholds that would call for a license amendment. Enclosure 1 to this paper presents the updated and refined ITAAC maintenance thresholds and scenarios when license amendments would be necessary for each threshold.

ITAAC Closure Verification Process

The staff has been developing an ICVP that it will use to determine whether ITAAC are properly completed based on a licensee's declarations in an ICL. The staff will apply the ICVP to verify that the requirements of 10 CFR 52.99 are correctly implemented. The staff has developed draft office instructions that delineate the ICVP and has initiated efforts to develop an information technology system that will facilitate implementation. The three major segments of the ICVP are (1) the ICL acceptance review, (2) the verification of ITAAC closure, and (3) the reevaluation and opening of a closed ITAAC. Enclosure 2 to this paper includes the current version of the ICVP flowchart.

During the acceptance review, the staff will determine whether an ICL has the correct format and references in accordance with the ICL templates provided in NEI 08-01, "Industry Guideline for the ITAAC Closure Process under 10 CFR Part 52," issued January 2009, and endorsed by the staff in Regulatory Guide 1.215. The templates were discussed in a series of public meetings that included NEI, industry, and other stakeholders.

The staff will verify that ITAAC are met based on a review of the information included in ICLs and the applicable inspection results documented in the Construction Inspection Program Information Management System (CIPIMS). NRO will lead this review, and will involve other offices such as the Office of Nuclear Security and Incident Response (NSIR) and Regional Offices, as needed. The ICVP will function similarly for the closure of ITAAC targeted for NRC inspection and nontargeted ITAAC.

The staff expects that closed ITAAC, which were maintained by licensee programs, will be reopened if the staff makes of an ITAAC finding[2] or if one of the four thresholds developed for ITAAC maintenance reporting requirements is exceeded. Enclosure 1 to this paper presents these thresholds.

The staff also issued Inspection Procedure 40600, "Licensee Program for ITAAC Management," which provides guidance to verify that licensees have established programmatic controls to govern the ITAAC closure process, including the process for preparing and approving closure letters. The guidance also verifies that the licensee has implemented an ITAAC maintenance program to ensure that SSCs continue to meet the acceptance criteria described in the ICLs until the Commission finding described in 10 CFR 52.103(g) is made.

The ICVP has received substantial review by stakeholders within the agency over the past year. Additionally, the NRO technical staff and Region II inspection staff evaluated the effectiveness of the process during a counterpart meeting in March 2010. During the meeting, the staff

[2] An ITAAC finding is a greater-than-minor inspection finding that occurs after the NRC receives the original ICL and directly affects the closure of an ITAAC.

evaluated six ITAAC closure scenarios and their expected outcomes. The staff confirmed that the ICVP incorporates the key elements to be considered in the process. The staff also plans to further test the ICVP during an ITAAC closure demonstration project with DOE and industry (described below). The ICVP is an essential NRC process that supports the Commission's finding in accordance with 10 CFR 52.103(g).

Simulated ITAAC Closure and Verification Demonstration Exercise

The staff is participating in a simulated ITAAC closure demonstration exercise with industry and DOE, the project sponsor. The purpose of the pilot exercise is to verify that both industry closure processes and NRC verification processes are reliable and efficient to support ITAAC closure. During this simulated exercise, Southern Nuclear Company and Westinghouse will prepare ITAAC closure packages and submit simulated 10 CFR 52.99(c)(1) notifications (ICL) to the NRC. The staff will have the opportunity to review these closure packages and completion letter submittals. The pilot exercise is expected to provide insights on the process and the expected level of detail and information contained in the closure packages and completion letters. Initially, the exercise will be limited to five ITAAC of varying complexity selected from the Westinghouse AP1000 design.

The exercise will be conducted under the assumptions that a COL has been issued with ITAAC and the plant is being constructed. Actual ITAAC performance by the licensee and inspections by Region II staff will not occur, but will be informed by elements of the inspection program. ITAAC performance and inspection data will be simulated to test the process.

The staff will facilitate the exercise through ITAAC public workshops, which should provide for an open and interactive series of discussions. Two meetings are currently planned to start the project, discuss observations on the exercise, identify issues with the processes, and provide solutions. Should the staff determine that there is value, DOE may expand the pilot to engage other new reactor vendors and applicants in similar demonstrations based on available resources, schedule implications, and anticipated benefits.

NRC Headquarters and Region II staff will participate by developing an inspection plan, documenting simulated inspection results in CIPIMS, evaluating the significance of inspection findings, exercising various aspects of the construction oversight process, and implementing the ICVP. One key objective for the exercise is to gain insights into communications among NRC Headquarters staff, Region II staff, and licensees if any issues are identified.

The exercise will also include an action to evaluate the surge in ITAAC closure submittals expected during the last year of a new nuclear power plant (NPP) construction project. Results from this evaluation may provide insights into how the NRC can better prepare for the expected impact of the surge on staff resources. At the completion of the exercise, the NRC staff will draft a lessons-learned report to highlight successes in the ICVP and to identify areas in need of continued refinement. The exercise is scheduled for completion by the end of February 2011. Enclosure 3 to this paper provides an abstract for this exercise.

RIS 2008-05 Update and Staff Training for the "Best Practices" in ITAAC Development

The NRC issued RIS 2008-05 in February 2008, (ADAMS Accession No. ML073190162) to communicate the best practices associated with the quality, clarity, and inspectability of ITAAC submitted as part of the applications for early site permits, standard DCs, or COLs. The NRC expects to issue Revision 1 to RIS 2008-05 in 2010, to expand discussion on the lessons learned and additional issues identified with respect to ITAAC inspectability.

Revision 1 to RIS 2008-05 will include a new "ITAAC scope" section. This section addresses issues identified during the staff's ongoing reviews of applications. The staff is resolving these issues with applicants through requests for additional information. The issues include considerations for additional ITAAC in design areas if needed, as well as wording which enhances the objectivity and clarity of ITAAC. The staff has also identified cases in which the inspections, tests, or analyses (ITA) and the acceptance criteria (AC) are inconsistent with each other and cases in which the ITA or AC do not align with the associated design commitment. RIS 2008-05, Revision 1 will also address the proper consideration of ITAAC revisions and the need for "extent-of-condition" evaluations for consistency and applicability issues of generic concerns.

The NRC has presented the issues included in Revision 1 to RIS 2008-05 during public workshops attended by industry and stakeholders. Workshop attendees mutually agreed that a need exists for an increased understanding by the NRC, industry, and the public on the meaning of certified ITAAC and on those ITAAC specific to a COL. The revision also reinforces the "best practices" for ITAAC approval and acceptance and the importance of submitting ITAAC suitable for inspection.

To complement the guidance provided in both RIS 2008-05 and its revision, training sessions were conducted for NRO technical reviewers and inspection program staff on April 20, 2010, and July 13, 2010. These training sessions summarized the lessons learned and discussed specific examples. The revision of RIS 2008-05 and the internal staff training sessions are helping to inform all stakeholders and to minimize recurrence of these types of issues.

Planned ITAAC Work

The staff plans continued interaction with industry and the public to further refine key elements of ITAAC closure and ITAAC maintenance. In addition, the staff has worked with other offices, such as NSIR, and has proactively incorporated specialized areas, such as security and EP into ITAAC maintenance guidance. On July 16, 2010, NEI submitted Revision 4 of NEI 08-01 (ML102010076) requesting NRC review and endorsement. The staff is reviewing this revision of NEI 08-01 and expects to issue a revision to Regulatory Guide 1.215 incorporating elements of ITAAC closure and ITAAC maintenance, as well as endorsing NEI 08-01 with any necessary clarifications.

The staff continues to work on issues involving Design Acceptance Criteria (DAC). The staff established a DAC working group in November 2009 to focus on issues associated with DAC resolution. The working group initiated and is currently participating in a DAC inspection to better develop the closure verification oversight process and details associated with digital instrumentation and controls DAC. Additionally, the working group continues to hold public

meetings with stakeholders, as well as briefings to the Advisory Committee on Reactor Safeguards.

COORDINATION:

This paper has been coordinated with the Office of General Counsel (OGC). OGC has no legal objection to this paper.

/RA/

Michael R. Johnson, Director
Office of New Reactors

Enclosures:
1. ITAAC Maintenance Thresholds
   and Associated License Amendments
2. ITAAC Closure Verification Process Flowchart
3. Simulated ITAAC Closure Demonstration Proposal

Inspections, Tests, Analyses, and Acceptance Criteria
Maintenance Thresholds and Associated License Amendments

Enclosure 1 to SECY 09-0119, "Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria," dated August 26, 2009, presented four thresholds for identifying when activities would materially alter the determination bases for inspections, tests, analyses, and acceptance criteria (ITAAC). Throughout the past year, the U.S. Nuclear Regulatory Commission staff refined the ITAAC maintenance thresholds after interactions with interested stakeholders during the ITAAC public workshop series. These refinements are intended to optimize the effectiveness of the thresholds and to clearly articulate the criteria for reporting. Each item below is an updated version of the thresholds proposed in Enclosure 1 to SECY 09-119. Following each threshold is a discussion on license amendments that would be necessary beyond the envelope of the threshold. These discussions describe scenarios that pertain to the threshold and state when a license amendment would be necessary.

Threshold 1: Postwork Verification

Will the postwork verification (PWV) use a significantly different approach than the original performance of the inspection, test, or analysis (ITA) as described in the original ITAAC notification?

Threshold 1 involves situations in which the occurrence of an event could call into question whether a licensee continues to meet an acceptance criterion (AC). Such situations could involve many types of maintenance activities, including component replacement. After work is complete, a PWV will be used to confirm that the licensee still meets the AC. The PWV is not a performance of the ITA because the licensee has already satisfied the requirement to perform the ITA; instead, the PWV and its results supplement the performance of the ITA to provide confidence that the licensee continues to meet the AC. The nature and the scope of the PWV will depend upon the nature of the initiating event, the maintenance activities undertaken, and the specific ITAAC that is implicated by the event. If the PWV represents an alternate approach that is significantly different from the approach described in the original ITAAC notification, a supplemental notification is necessary to provide the agency and members of the public information that is material to the agency's determination on ITAAC.

Because the PWV is not a performance of the ITA but rather a supplement to the performance of the ITA, the PWV does not have to comport with the ITA set forth in the license. However, the licensee would need to seek an amendment to that ITA in the license if no reasonable "alternate" PWV approach is available to demonstrate that the AC continues to be met. Whether an alternative PWV is reasonable or not depends on several factors, including the engineering justification provided and the wording of both the ITA and the AC. A reasonable alternative to the original ITA represents a different, yet acceptable, engineering equivalent for performing the activity prescribed in the ITAAC. As an example, if a test was the original prescribed ITA, then the PWV should also be a test, or possibly a combination of a test and analysis or a test and an inspection. The PWV methodology should generally follow the methodology used in the original prescribed ITA.

A license amendment would also be necessary if the PWV reveals that the licensee never met the AC because the original ITA, as worded in the license, was flawed.

Threshold 2: Engineering Changes

Will an engineering change be made that materially alters the determination that the acceptance criteria are met?

License amendments would also be necessary if the engineering change results in the need to identify new AC or if the engineering change results in a design for which the AC as written cannot be demonstrated using the original ITAs.

Threshold 3: Population of Systems, Structures, and Components

Will there be additional items that need to be verified through the ITAAC?

A license amendment would be needed if there are additional items subject to verification through the existing ITAAC, but the licensee proposes not to perform the ITAs specified in the ITAAC. An amendment would also be required if new or amended ITAAC are needed to cover new items (e.g., the new items are of a different type than those covered in the original ITAAC).

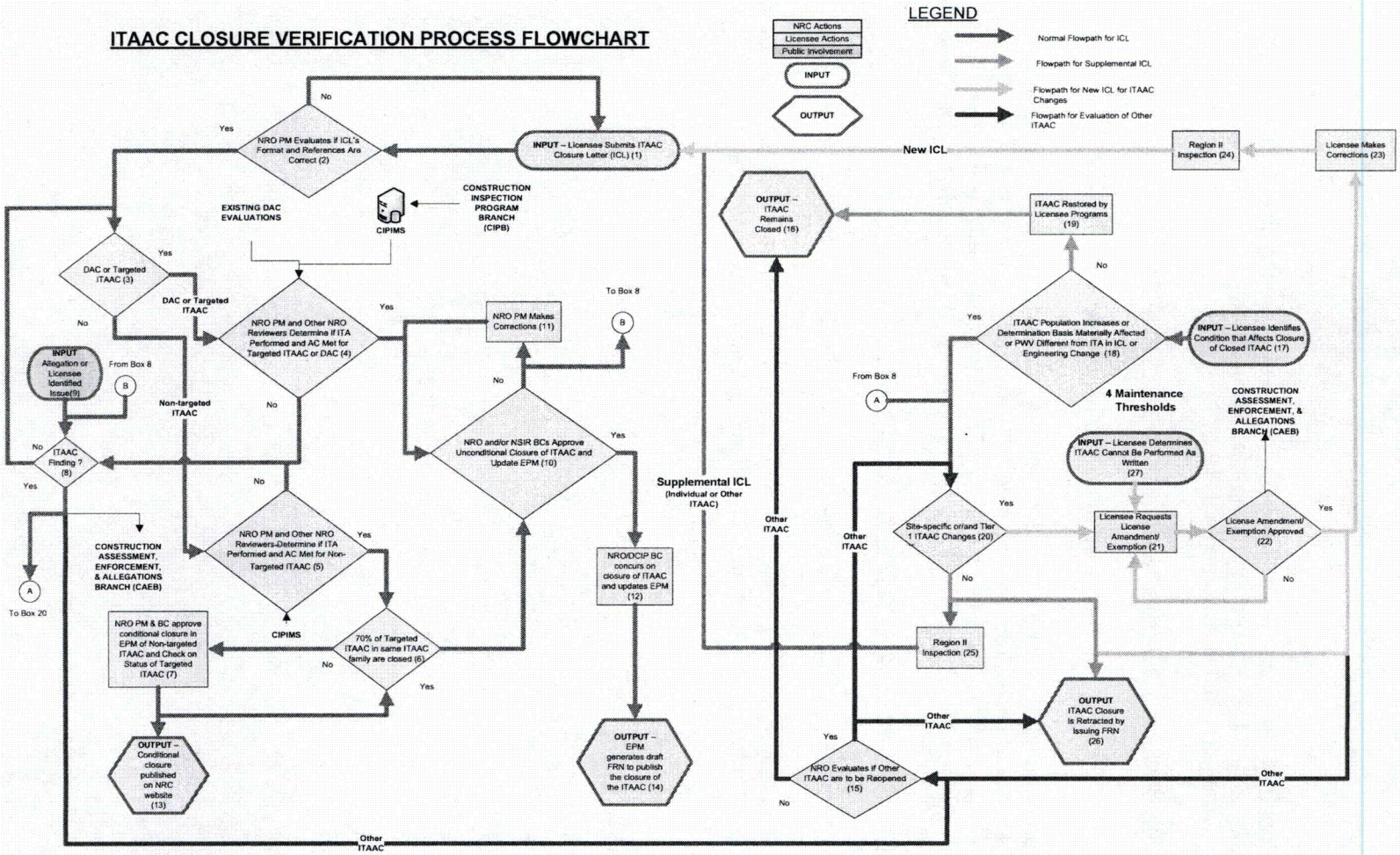
Threshold 4: Complete and Valid ITAAC Representation

Will any other licensee activities materially alter the ITAAC determination basis?

A license amendment would be needed if an update of the determination basis necessitates a change to any portion of ITAAC in the license for reasons not covered under thresholds 1, 2, and 3.

DRAFT COPY

ITAAC CLOSURE VERIFICATION PROCESS FLOWCHART



Enclosure 2

DESCRIPTION OF ITAAC CLOSURE VERIFICATION PROCESS

(1) Licensee may submit to the NRC initial, new, or supplemental ICLs. If (22) is "yes", the licensee submits a new ICL for a license amendment/exemption for changes to site-specific and/or Tier 1 ITAAC. If (20) is "no", the licensee submits a supplemental ICL for licensee actions that may affect the closure of a previously closed ITAAC, physical installations, post-maintenance verification of SSCs associated with an ITAAC, and the content of an ICL previously submitted to the NRC. NRO PM determines during an "acceptance review" if the ICL has correct format and references based on the examples of ICLs developed by the NRC and industry.

(2) If (2) is "no", either the format or/and the references of an ICL is/are not correct. The NRC informs the licensee by a letter of the rejection of the ICL. Licensee corrects the errors in the ICL and resubmits it for "acceptance" by the NRC. If (2) is "yes", both the format and the references of an ICL are correct. The DCIP PM should also verify that all ITAAC-related construction findings for an ITAAC or DAC are closed. The NRO PM manually acknowledges acceptance of the ICL in EPM by an electronic signature.

(3) NRO PM determines if the ITAAC is a DAC or Targeted ITAAC.

(4) If (3) is "yes", the ITAAC is either a DAC or targeted ITAAC. NRO PM determines for this DAC or ITAAC if its ITA was performed and its AC was met based on sufficient information in the ICL. This decision is made by referring to inspection records in CIPIMS, licensee's certifications in the ICL, and if necessary, licensee's supporting documentation at plant site. For the DAC or targeted ITAAC, primary and secondary technical reviewers, assigned to this review by their NRO and/or NSIR respective branch chiefs, may provide input to the decision on the closure of the DAC or ITAAC. The inspection records for the DAC will contain evaluations of the portion of the design not reviewed during the design reviews for the design certification application. NRO PM and reviewers approve the DAC or targeted ITAAC as being ready for closure and manually indicate their approval in EPM by their electronic signatures.

(5) If (3) is 'no', the ITAAC is a non-targeted ITAAC. NRO PM and reviewers determine for the non-targeted ITAAC if its ITA was performed and its AC met based on sufficient information and the licensee certifications in the ICL This review will be similar to that for targeted ITAAC and DAC, except there may be no inspection records in CIPIMS to further corroborate the closure of a non-targeted ITAAC.

(6) If (5) is "yes", NRO PM will determine if 70% of targeted ITAAC in the same ITAAC family, as non-targeted ITAAC, are closed.

(7) If (6) is "no", the NRO PM & BC conditionally close the non-targeted ITAAC in EPM. The conditional closure of a non-targeted ITAAC does not go through entire ITAAC review cycle, but the review stops at a designated NRO BC. The conditional closure can be revoked by inspection, allegations, maintenance, or design issues that arise against ITAAC in the same ITAAC family as the non-targeted ITAAC. If non-targeted ITAAC is conditionally closed, NRO PM manually checks periodically on whether the 70% of targeted ITAAC in same ITAAC family have been closed based on information in EPM concerning the targeted ITAAC. If the non-targeted ITAAC is conditionally closed, the NRC will publish its conditional closure on an NRC website selected for that purpose. If the conditional closure of the non-targeted ITAAC is revoked, that will be published on this same website.

(8) If either (4) or (5) or (10) is/are "no" or if there is an allegation or licensee-indentified issue (9) that affects ITAAC closure, NRO PM informs the DCIP assessment branch about the ITA not being performed or/and the AC not being met. DCIP assessment team, Region II, and other NRO and/or NSIR divisions will determine if there should be an ITAAC finding. An ITAAC finding will prevent an ITAAC from being closed and could cause a closed ITAAC and other closed ITAAC in same ITAAC family to be reopened. If an ITAAC finding requires ITAAC changes, a new ICL(s) is/are required at (22), but if changes are only to physical installation or/and documentation supporting closure of an ITAAC in an ICL or at site, then supplemental ICL(s) will be required at (20). If (8) is "no", then ITAAC re-enters ITAAC review process just prior to (3).

(9) Allegation or licensee-identified issue against an open or previously closed ITAAC that affects its closure is received and reviewed by DCIP, Region II, applicable NRO and/or NSIR divisions. Allegation may result in an ITAAC finding at (8) which may (a) prevent the closure of an ITAAC, or (b) cause a closed ITAAC and other ITAAC in same ITAAC family to be reopened.

(10) If (4) or (6) is "yes", NRO and/or NSIR branch chiefs manually approve the DAC or targeted ITAAC or non-targeted ITAAC for unconditional closure in EPM by their electronic signatures.

(11) If (10) is "no", NRO and/or NSIR BCs have made comments which have to be corrected before the ITAAC can be approved for closure. A NRO PM will make the changes, and they will be reviewed by a designated NRO BC for concurrence. If NRO and/or NSIR BCs have comments that call for some action by the licensee, the NRO PM will initiate contact with the licensee based on direction by NRO/BC.

(12) If (10) is "yes," DCIP/BC concurs with closure of the ITAAC and issuance of the FRN. DCIP/BC's concurrence is manually input into EPM

(13) NRO PM publishes conditional closure of the non-targeted ITAAC on an NRC website.

(14) EPM generates a draft FRN when DCIP/BC concurs on closure of the ITAAC. The draft FRN is sent to appropriate NRC group for issuance. The draft FRN may announce the closure of one or several ITAAC.

(15) When an ITAAC cannot be closed "yes" output at (8) or must be reopened "yes" output at (22), NRO evaluates if the closures of other ITAAC in the same ITAAC family are affected. A supplemental ICL must go through the process for ITAAC closure before the impact on other closed ITAAC in same ITAAC family can be determined. If (15) is no, then the other ITAAC being evaluated remain closed (16).

(16) ITAAC remains closed if licensee through its programs restores ITAAC to compliant condition (19). In addition, if (15) is "no", then ITAAC remain(s) closed (16).

(17) Licensee actions precipitate a condition that affects the closure of a previously closed ITAAC.

(18) Licensee determines one or more of the four maintenance thresholds for an ITAAC has been exceeded due to one or more of the following: (a) population of SSCs identified in ITAAC has increased or (b) the determination basis of the ICL for ITAAC is materially affected due to licensee activities, or c) post-work verification, different from the original ITA for ITAAC, is performed which requires an engineering justification, or (d) an engineering change for SSCs associated with the ITAAC materially affects the determination that the acceptance criteria of the ITAAC was met..

(19) If (18) is "no", then licensee programs, like PI&R, configuration management, etc, restore ITAAC to compliant condition, and ITAAC remains closed (16).

(20) If (8), (15), or/and (18) is "yes", then the licensee evaluates if changes are required to Tier 1 or site-specific ITAAC due to (a) exceeding a maintenance threshold (18) , (b) ITAAC finding (8), and/or (c) for other ITAAC being reopened (15).

(21) If (20) is a "yes", then the licensee requests a license amendment and exemption to seek changes to a Tier1 ITAAC or submits just a license amendment to seek changes to a site-specific ITAAC.

(22) License amendment approved or disapproved by the NRC. If approved, the closure of ITAAC is affected because of Tier 1 or/and site-specific ITAAC changes, and the licensee submits new ICL(s). If not approved, then the licensee will make changes to license amendment and resubmit to the NRC. Other closed ITAAC in same ITAAC family as ITAAC being reopened are evaluated to determine if their closure is also affected (15). The determination will have to made for those other ITAAC if they are to reopened also.. The NRO PM shall status the individual and other ITAAC, which is/are contained in license amendment, as being "reopened" in EPM when license amendment is approved. For those ITAAC that need to be reopened, EPM will be updated to reflect the status of those ITAAC as being "reopened."

(23) Licensee receives a letter from NRC about the license amendment being approved. Licensee makes the necessary corrections which consist of any or all of the following for all the ITAAC affected: (a) modifications to affected physical installations, (b) change(s) to ITAAC and/or Tier 1 of Design Control Document of specific certified design, (c) revision to supporting calculations and analyses for an ITAAC, (d) performance of a new ITA, (e) validation that new AC of affected ITAAC is/are now met, and/or (f) changes to content of ICL(s). The other ITAAC in same ITAAC family go through this same process when a licensee amendment for singular ITAAC is approved.

(24) Region II performs additional inspections, as required, after informal notice from licensee to verify the licensee corrections identified in Item (22) above. After Region II performs inspections, if necessary, then the licensee makes any additional changes and submits new ICL(s) to the NRC because of the ITAAC changes identified for the individual and other ITAAC.

(25) If (20) is "no", then Region II performs additional inspections, as required, after informal notice from licensee to verify the licensee corrections made because one or more of the four maintenance thresholds was crossed.. After Region II performs inspections, if necessary, then the licensee makes any additional changes and submits a supplemental ICL to the NRC. The supplemental ICL should indicate whether the closure of the ITAAC is affected or not. If closure is affected, the NRC will issue an FRN (26) to revoke its closure. The determination of whether closure of ITAAC is affected may be made by the NRC after review of supplemental ICL or before supplemental ICL is submitted by the licensee. If the determination is made after review of supplemental ICL, then a revised supplemental ICL will have to submitted by the licensee that reflects that closure of ITAAC is affected.

(26) If (15) or (22) is "yes", then an FRN is published to revoke the closure of the ITAAC previously closed.

(27) Licensee determines that a given ITAAC as written cannot be performed because ITA cannot be implemented or AC met.

U.S. Nuclear Regulatory Commission
U.S. Department of Energy
Southern Nuclear Company
Westinghouse

Simulated Inspections, Tests, Analyses, and Acceptance Criteria Closure and Verification Demonstration Proposal

July 29, 2010

Enclosure 3

Background:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," requires combined license (COL) applicants to submit and perform inspections, tests, analyses, and acceptance criteria (ITAAC) in order to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the COL. The licensee notifies the U.S. Nuclear Regulatory Commission (NRC) that it has completed the ITAAC by submitting an ITAAC closure letter (ICL) stating that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. In turn, the NRC staff will review all ICLs to verify that the prescribed ITAAC are met, and will then issue a *Federal Register* notice (FRN) of the staff's determination of the successful completion of the ITAAC. When the staff has verified that all ITAAC have been closed, the Commission will decide whether to make a finding that the licensee has met the acceptance criteria in the COL. If the Commission finds that all the acceptance criteria are met, then the licensee may operate the facility under 10 CFR 52.103(g).

Currently, the staff is gaining experience by inspecting the ITAAC contained in the Vogtle limited work authorization and by conducting a pilot inspection of design acceptance criteria related to the South Texas Project. Additionally, the staff has several initiatives in progress to ensure that the ITAAC closure and verification processes are effective and efficient. These initiatives include: (1) holding NRC internal workshops to develop the inspection strategy and exercise the ITAAC closure verification process, (2) working with applicants on the development and refinement of the ITAAC schedule, (3) preparing proposed rule language for ITAAC maintenance, (4) holding bimonthly Category 3 public workshops with stakeholders to evaluate and resolve issues associated with ITAAC closure, and (5) revising Regulatory Issue Summary 2008-05, "Lessons Learned To Improve Inspections, Tests, Analyses, and Acceptance Criteria Submittal," dated February 27, 2008, on ITAAC quality and inspectability and conducting related internal training.

In addition to the ongoing activities, the U.S. Department of Energy (DOE) proposes to sponsor an exercise with the NRC and industry to demonstrate the review and closure of ITAAC. This exercise can be quite valuable because previous commercial reactors that were constructed and licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," did not use ITAAC. Furthermore, industry and the NRC staff have long known that a substantial percentage of ITAAC will be completed in the year leading to the scheduled date for fuel loading. Accordingly, through discussions with DOE on its proposal, the staff suggested a study on the expected surge in ITAAC submittals during the last stages of construction to evaluate possible strategies to effectively and efficiently complete the reviews at the end of the construction process. The staff and DOE have agreed to cooperatively pursue this demonstration project. DOE will focus its efforts on the selection and the possible financial reimbursement of participating licensees and vendors. The NRC staff will facilitate the ITAAC closure process, including the coordination of efforts with the participating licensees and vendors.

Objective and Approach:

The objective of this demonstration project is to verify that both the industry closure processes and the NRC verification processes are reliable and efficient to support ITAAC closure. Specifically, the industry will simulate the development of several ITAAC closure documents and the submission of the associated ITAAC closure notifications under 10 CFR 52.99(c)(1). During this process, NRC Region II staff will simulate inspection planning and the documentation of inspection results in the Construction Inspection Program Information Management System (CIPIMS). NRC Headquarters staff in the Office of New Reactors (NRO) will simulate the review of ICLs submitted by the applicant and inspection results documented in CIPIMS. NRO will also simulate the NRC's internal ITAAC closure verification process.

Participants in this exercise will initially include the NRC, DOE as a project sponsor, and Westinghouse and Southern Nuclear Company (SNC) as the participating applicants. Based on available resources, schedule implications, and expected benefits, the NRC staff may engage other new reactor vendors and applicants in similar demonstrations.

Lastly, the exercise will involve evaluation of the surge in ITAAC closure submittals expected during the last year of construction of a new nuclear power plant.

For purposes of this demonstration project, participants will assume that ITAAC exist and that the plant is under construction. Actual NRC Region II inspections will not take place; instead, inspection data will be simulated to test the process.

Demonstration Plan:

The ITAAC closure demonstration project will include the following five ITAAC from the AP1000, Revision 17 design:

(1) ITAAC 2.1.02.07a.i – The Reactor Coolant System (RCS) Harsh Environment Type Test

(2) ITAAC 2.2.01.04a.ii – Containment System Impact Testing

(3) ITAAC 2.2.02.01 – Passive Containment Cooling Functional Arrangement

(4) ITAAC 2.2.03.08c.i – Injection Line Flow Resistance Testing and Analysis

(5) ITAAC 2.6.03.08 – DC System Fault Current Analysis

Westinghouse and SNC will participate in the initial demonstration. Other design centers and license applicants may participate in future demonstrations. The exercise will be facilitated using the existing NRC ITAAC workshop infrastructure, which is open to public participation and provides for a series of open and interactive discussions as the exercise progresses. The project is divided into the following four stages:

(1) ITAAC Performance

The applicant will simulate the performance of the selected ITAAC and develop the documentation required to support ITAAC closure. As part of the ITAAC closure package, the applicant will prepare the ICL to provide information sufficient to demonstrate that the inspections, tests, and analyses have been performed and that the acceptance criteria are met based on the templates provided in NEI 08-01, "Industry Guideline for the ITAAC Closure under 10 CFR Part 52," issued January 2009.

Concurrently, NRC Region II staff will prepare an inspection plan for the selected ITAAC, document the selected ITAAC information and the simulated inspection results in CIPIMS, and generate information reports pertaining to ITAAC inspections. Stage 1 of the exercise concludes when the ITAAC performance demonstration is completed and when the ITAAC closure package is prepared and made available to the NRC.

(2) ITAAC Closure

Once the ITAAC closure package is made available to the NRC and the agency receives the ICL, the staff will exercise the NRC's ITAAC closure verification process. The NRO staff will process the ICL as outlined in the draft ITAAC closure verification process office instruction and its appendices. This review may include the use of NRO technical staff to evaluate complex technical information. Stage 2 of the exercise concludes when the staff verifies proper ITAAC closure and simulates the publishing of an FRN.

(3) Exercise Workshop

The NRC will hold a public workshop to summarize and discuss the exercise and to present the ICLs. Participants will discuss their observations of the exercise, identify issues with the process, and propose solutions. Westinghouse will also present the analysis results of the makeup and volumes of the system-specific ITAAC in the expected surge of ITAAC closure submittals during the last year of construction. The issues associated with the expected surge in ITAAC should be discussed in detail to identify strategies to minimize any schedule impact. Stage 3 of the exercise concludes when the participants complete and achieve the goals of the workshop.

(4) Lessons Learned

The NRC will draft a lessons-learned report that highlights successes in the ITAAC closure and verification processes and that details areas that could be further refined. The report will include input gathered from participants and the public throughout the exercise and during the public workshops. The staff will continue to coordinate with the applicant on the refinement of the ITAAC closure schedule, based on insights that it

obtains from the analysis of the ITAAC closure surge, to mitigate any potential delays in ITAAC inspections and closures and to minimize the impact on NRC resources.

Milestones Summary:

Milestones	Date
Project Development Meeting	Completed
Abstract Development	Completed
NRO Management Endorsement	Completed
DOE/Westinghouse/SNC Endorsement	Completed
Public Meeting/Project Initiation	Completed
Stage 1 Complete (ITAAC Performance)	September 30, 2010
Stage 2 Complete (ITAAC Closure)	November 19, 2010
Stage 3 Complete (Exercise Workshop)	December 16, 2010
Stage 4 Complete (Lessons Learned)	February 28, 2011

Points of Contact:

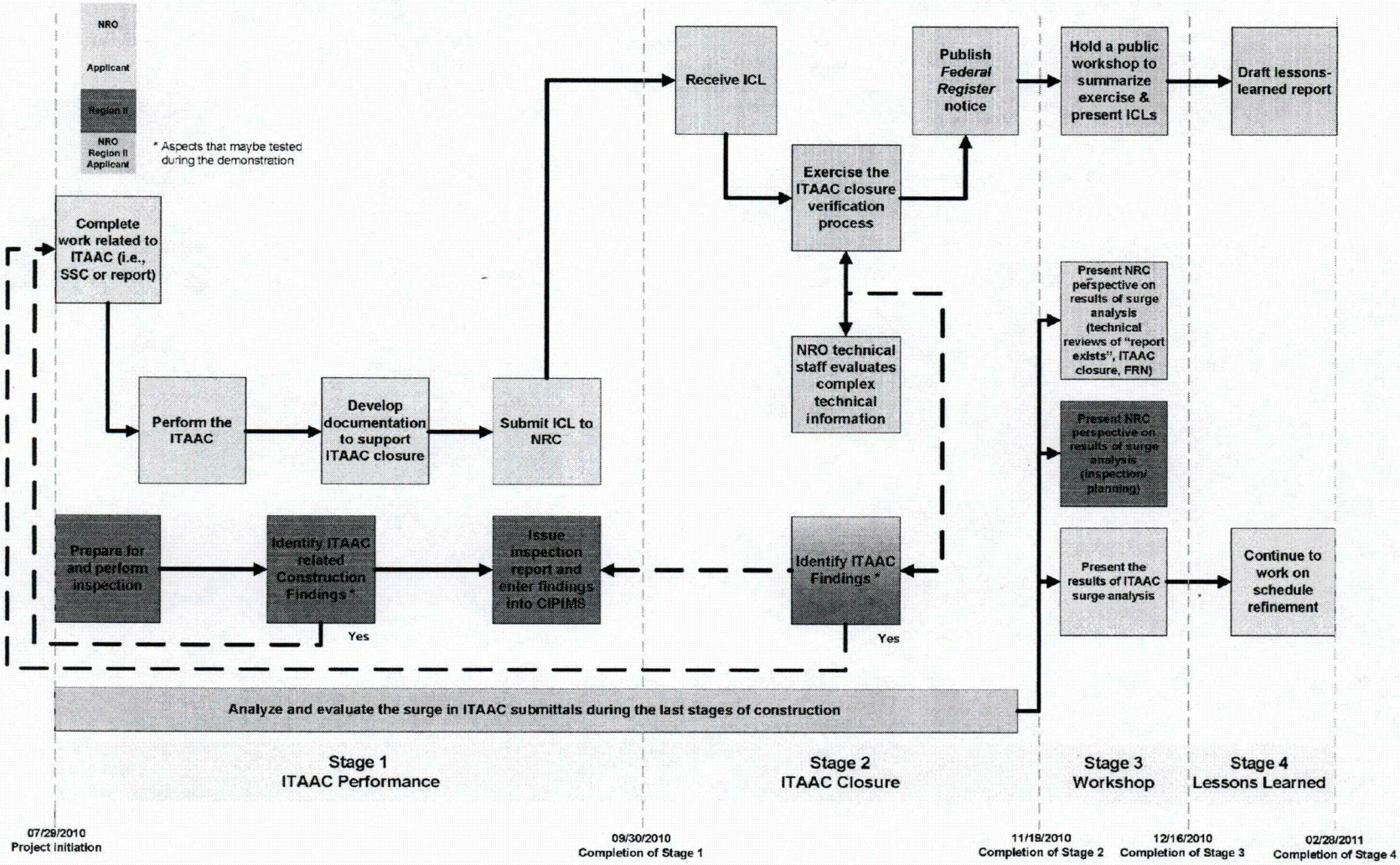
Name	Organization	E-mail Address
Tom Miller	DOE, NE-31	tom.miller@nuclear.energy.gov
Mark Kowal	NRC/DCIP	mark.kowal@nrc.gov
Richard Laura	NRC/DCIP	richard.laura@nrc.gov
Jim Gaslevic	NRC/DCIP	james.gaslevic@nrc.gov
MJ Jardaneh	NRC/DCIP	mahmoud.jardaneh@nrc.gov
Juan Peralta	NRC/DCIP	juan.peralta@nrc.gov
Alan Blamey	NRC Region II	alan.blamey@nrc.gov
Chuck Pierce	SNC	crpierce@southernco.com
Brandon Waites	SNC	bwwaites@southernco.com
Robert Sisk	Westinghouse/HQ	sisk1rb@westinghouse.com
Thom Ray	Westinghouse/HQ	raytj@westinghouse.com

Exercise Guidance:

- 10 CFR Part 52
- NRC Inspection Manual Chapter 2503, "Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)," dated October 3, 2007
- ITAAC Inspection Procedures 65000 series
- Regulatory Guide 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52"
- NEI 08-01
- NRC ITAAC Closure Verification Process Office Instructions (Draft)

NRO/DCIP/CIT
July 27, 2010

Simulated ITAAC Closure and Verification Demonstration Flowchart



RULEMAKING ISSUE
NOTATION VOTE

August 30, 2010 SECY-10-0117

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: PROPOSED RULE: REQUIREMENTS FOR MAINTENANCE OF INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (RIN 3150-AI77)

PURPOSE:

The purpose of this paper is to request Commission approval to publish for public comment a proposed rulemaking that would amend requirements related to verification of nuclear power plant construction activities through inspections, tests, analyses, and acceptance criteria (ITAAC) under a combined license.

SUMMARY:

The U.S. Nuclear Regulatory Commission (NRC) staff seeks Commission approval of proposed amendments to the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.99, "Inspection during construction," related to verification of nuclear power plant construction activities through ITAAC under a combined license. Specifically, the staff proposes new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure letter. The new provisions would require licensees to (1) report new information materially altering the basis for determining that a prescribed inspection, test, or analysis was performed as required, or finding that a prescribed acceptance criterion is met, (2) document the basis for all ITAAC notifications, and (3) notify the NRC of completion of all ITAAC activities.

CONTACT: Nanette Gilles, NRO/DNRL
301-415-1180

In addition, the staff proposes corrections to existing language in 10 CFR 2.340, "Initial decision in certain contested proceedings; immediate effectiveness of initial decisions; issuance of authorizations, permits, and licenses," and 10 CFR 52.99 to correct errors and clarify ambiguous language and to make it consistent with language in the Atomic Energy Act of 1954, as amended (AEA).

BACKGROUND:

When the Commission first issued 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (54 FR 15371; April 18, 1989), it included 10 CFR 52.99 to make it clear that the inspection carried out during construction under a combined license would be based on ITAAC proposed by the applicant, approved by the staff, and incorporated in the combined license. At that time, the Commission made it clear that, although 10 CFR 52.99 envisioned a "sign-as-you-go" process, in which the NRC staff would sign off on inspection units and notice of the staff's sign-off would be published in the *Federal Register*, the Commission itself would make no findings on construction until construction was complete.

In 2007, the Commission revised 10 CFR Part 52 to enhance the NRC's regulatory effectiveness and efficiency in implementing its licensing and approval processes (72 FR 49351; August 28, 2007). In that revision, the NRC amended 10 CFR 52.99 to require licensees to notify the NRC that the prescribed inspections, tests, and analyses in the ITAAC have been completed and that the acceptance criteria have been met. The revision also requires that these notifications contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. The NRC added this requirement to ensure that combined license applicants and holders were aware that it was the licensee's burden to demonstrate compliance with the ITAAC and the NRC expected the notification of ITAAC completion to contain more information than just a simple statement that the licensee believes the ITAAC had been completed and the acceptance criteria met.

The notifications currently required by 10 CFR 52.99(c)(1) perform two functions, as discussed in the supplementary information for the 2007 final rule amending 10 CFR Part 52 (72 FR 49352; August 28, 2007, at 49450 (second column)). First, the notifications alert the NRC to the licensee's completion of the ITAAC[1] and ensure that the NRC has sufficient information to complete all of the activities necessary for the Commission to find whether all of the ITAAC acceptance criteria have been or will be met before initial operation (the "will be met" finding is relevant to any hearing on ITAAC under 10 CFR 52.103, "Operation under a combined license"). Second, the notifications ensure that interested persons will have access to information on both completed and uncompleted ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing on acceptance criteria.

After completing the 2007 rulemaking, the staff began developing guidance on the ITAAC closure process and the requirements under 10 CFR 52.99. In October 2009, the NRC issued regulatory guidance for the implementation of the revised 10 CFR 52.99 in Regulatory Guide (RG) 1.215, "Guidance for Closure Under 10 CFR Part 52." This regulatory guide endorsed

[1] In this discussion, the phrases "completion of the ITAAC" and "ITAAC completion" mean that the licensee has determined that (1) the prescribed inspections, tests, and analyses were performed and (2) the prescribed acceptance criteria are met.

guidance developed by the Nuclear Energy Institute (NEI) in NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3, issued January 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090270415).

After considering information presented by industry representatives in a series of public meetings, the staff realized that some additional implementation issues were left unaddressed by the various provisions in 10 CFR Part 52. In particular, the staff determined that the combined license holder should provide additional notifications to the NRC following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1). The staff refers to the time after this ITAAC closure notification but before the date the Commission makes the finding under 10 CFR 52.103(g) as the "ITAAC maintenance period."

Since mid-2008, the staff has held numerous meetings that have discussed the topic of ITAAC maintenance. In SECY-09-0119, "Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria," dated August 26, 2009, the staff informed the Commission of its progress in resolving issues concerning ITAAC maintenance and reporting, including the threshold for notification of events that may result in the acceptance criteria of successfully completed ITAAC no longer being satisfied. That paper described new types of notifications that licensees may make to the NRC to address instances when licensee activities affect previously completed ITAAC after the licensee submits an ITAAC closure letter to the NRC. The staff stated in the paper that it planned to propose that the Commission supplement 10 CFR Part 52 to include additional notification requirements to address ITAAC maintenance. The staff reiterated its plans to propose rulemaking on ITAAC maintenance in a September 22, 2009, Commission meeting on this topic.

More recently, the staff held two public meetings in March 2010 to discuss draft proposed rule text that it made available to the public in February 2010. The staff considered feedback from external stakeholders during those meetings in its development of this proposed rule. In addition, NEI submitted written comments on the staff's plans to amend 10 CFR 52.99 in a letter dated April 29, 2010 (ADAMS Accession No. ML101300103), reiterating many of the comments made by NEI representatives in the March 2010 public meetings. To maintain the schedule for this proposed rulemaking, the staff responded to NEI on June 21, 2010 (ADAMS Accession No. ML101590526) by requesting that it resubmit those written comments from its letter that NEI believes are applicable to the published proposed rule number. At that time, the NRC staff will address NEI's comments, as part of the NRC's response to all of the comments received by the NRC on the proposed rule.

In parallel with the discussions on the draft proposed rule, NEI has been revising its ITAAC closure guidance in NEI 08-01 to address the topic of ITAAC maintenance. On July 16, 2010, NEI submitted Revision 4 of NEI 08-01 for NRC review and endorsement (ADAMS Accession No. ML102010076). The staff is reviewing this latest revision of NEI 08-01 and expects to issue a revision to RG 1.215 for public comment, endorsing NEI 08-01 and providing any necessary clarifications, by the end of 2010.

As stated in the Commission's staff requirements memorandum (SRM) SRM-M091208, dated January 13, 2010, the staff considered how the concept of aggregate impact may apply to this proposed rule. Consistent with the staff's plans to address aggregate impacts in rulemaking, the staff has had significant interaction with external stakeholders during development of the

proposed rule and on draft guidance to support the rule. The staff plans to issue draft regulatory guidance shortly after publication of the proposed rule. In addition, in March 2010, the staff issued Inspection Procedure 40600, "Licensee Program for ITAAC Management," that provides guidance to verify licensees have implemented ITAAC maintenance programs to ensure that structures, systems, and components continue to meet the ITAAC acceptance criteria until the Commission makes the finding under 10 CFR 52.103(g) allowing operation. The staff expects that all guidance necessary to implement this rule will be available at the time that the final rule becomes effective.

This rule is not expected to result in any aggregate impact in implementation because the rule's requirements apply during construction only and do not affect combined license issuance. In addition, prospective licensees that would be subject to the new requirements will have significant exposure to them well before the time the requirements will become effective and, therefore, will have adequate time to determine how to implement the rule. Finally, the issues involving ITAAC maintenance are fundamental to the process for the Commission's finding under 10 CFR 52.103(g), so the staff does not have latitude in determining the implementation period of the rule. Therefore, the staff concludes that no additional action to address the concept of aggregate impact is necessary.

DISCUSSION:

The staff proposes the following new notifications after ITAAC closure:

- Notification of new information on ITAAC closure
- Supplemental ITAAC closure notification
- All ITAAC complete notification

In general, the reasons for these proposed new notifications are analogous to the reasons presented in the 2007 rulemaking for the existing 10 CFR 52.99(c) notifications: to ensure that the NRC has sufficient information, in light of new information developed or identified after the ITAAC closure notification under 10 CFR 52.99(c)(1), to complete all of the activities necessary for the Commission to make a finding on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing. The following sections of this paper describe each of the proposed notification and documentation requirements in this rulemaking, and the bases for each of the proposed requirements.

Notification of New Information on ITAAC Closure

The licensee is responsible for maintaining the validity of the ITAAC conclusions after completion of the ITAAC. If the ITAAC determination basis is *materially altered*, the staff believes that the licensee should be required to notify the NRC. Through public workshops and stakeholder interaction, the NRC has developed thresholds to identify when activities would materially alter the basis for determining that a prescribed inspection, test, or analysis was performed as required, or finding that a prescribed acceptance criterion is met (the "ITAAC determination basis"). One obvious case is that proposed 10 CFR 52.99(c)(3)(i) would require a notification to correct a material error or omission in the original ITAAC closure letter.

Section 52.6, "Completeness and accuracy of information," paragraph (a), requires that information provided to the Commission by a licensee be complete and accurate in all material respects. However, it might be the case that the original closure notification was complete and accurate when sent, but subsequent events materially alter the ITAAC determination bases. Also, a material error or omission might not be discovered until after the ITAAC closure notification is sent. It is possible that new information materially altering the ITAAC determination bases would not rise to the reporting threshold under 10 CFR 52.6(b). As required by 10 CFR 52.6(b), licensees must notify the Commission of information identified by the licensee as having, for the regulated activity, a significant implication for public health and safety or the common defense and security.

Given the primary purpose of ITAAC—to verify that the plant has been constructed and will be operated in compliance with the approved design—the NRC believes that it cannot rely on the provisions in 10 CFR 52.6 for licensee reporting of new information materially altering the ITAAC determination bases. The reasons for this conclusion are as follows:

1. Material errors and omissions in ITAAC closure notifications, relevant to the accuracy and completeness of the documented bases for the Commission's finding on ITAAC, may nonetheless be determined in isolation by a licensee as not having a significant implication for public health and safety or common defense and security.

2. A Commission finding of compliance with acceptance criteria in the ITAAC is required, under Section 185.b of the AEA, in order for the combined license holder to commence operation.

3. The addition of specific reporting requirements addressing information relevant and material to the ITAAC finding ensures that NRC will get the necessary reports as a matter of regulatory requirement, and allows the NRC to determine the timing and content of these reports so that they serve the regulatory needs of the NRC.

Therefore, the staff recommends requiring that such issues will be reported under proposed 10 CFR 52.99(c)(3)(i). In addition to the reporting of material errors and omissions, the staff has identified other circumstances in which reporting under this provision would be required (i.e., reporting thresholds). The staff described these reporting thresholds in SECY-09-0119 and discusses them at a high level in Section IV of Enclosure 1.

Proposed 10 CFR 52.99(c)(3)(i) would require licensees to notify the NRC of new information materially altering the basis for determining that an inspection, test, or analysis was performed as required, or finding that an acceptance criterion is met. Licensees would be required to notify the NRC no later than 7 days after the licensee has determined that the new information materially alters the ITAAC determination basis. The purpose of this initial notification would be to alert the NRC staff responsible for oversight of construction inspection activities to the fact that additional activities may be scheduled that affect a structure, system, or component (including physical security hardware) or program element for which one or more ITAAC have been closed. This would allow the NRC inspection staff to discuss the licensee's plans for resolving the issue and determine if the staff wants to observe any of the upcoming activities for the purpose of making a future staff determination about whether the acceptance criteria for those ITAAC continue to be met.

Supplemental ITAAC Closure Notification

The second new notification is contained in proposed 10 CFR 52.99(c)(3)(ii). When making the 10 CFR 52.103(g) finding, the Commission must have information sufficient to determine that the relevant acceptance criteria are met despite the new information supplied in the notification under 10 CFR 52.99(c)(3)(i). The licensee's concise statement of the bases for resolving the issue, which was the subject of the new information notification, discussion of any action taken, and a list of the key licensee documents supporting the resolution and its implementation, would assist the NRC in making its independent evaluation of the issue. Apart from the NRC's use of the information, the staff also believes that making this information available to the public is necessary to ensure that interested persons will have sufficient information to review when preparing a request for a hearing under 10 CFR 52.103, comparable to the information provided under 10 CFR 52.99(c)(1), as described in the statements of consideration for the 2007 10 CFR Part 52 rulemaking (72 FR 49352; August 28, 2007).

Accordingly, the staff proposes that, after a licensee makes a new information notification under 10 CFR 52.99(c)(3)(i), it be required to submit what is essentially a resolution notification to the NRC in the form of a supplemental ITAAC closure letter. This supplemental ITAAC closure notification, described in proposed 10 CFR 52.99(c)(3)(ii), would require the licensee to submit a written notification of the resolution of the circumstances that prompted the 10 CFR 52.99(c)(3)(i) new information notification. The supplemental ITAAC closure letter must contain sufficient information demonstrating that, notwithstanding the information that prompted the 10 CFR 52.99(c)(3)(i) notification, the prescribed inspections, tests, and analyses have been performed as required and the prescribed acceptance criteria are met. Supplemental ITAAC closure letters should explain the need for the supplemental letter, outline the resolution of the issue, and confirm that the ITAAC acceptance criteria continue to be met. Supplemental ITAAC closure letters must include a level of detail similar to the level of information required in initial ITAAC closure letters under 10 CFR 52.99(c)(1).

The staff proposes that the supplemental ITAAC closure letter be submitted within 30 days after licensee resolution of the issue. The staff selected the 30-day period as a reasonable compromise between the competing goals of prompt notification and the recognition that documentation of the resolution may be quite extensive, depending on the significance and number of activities necessary for the licensee to resolve the issue. The staff proposes that NRC receipt of the notification be published in the *Federal Register*, at least until the last date for submission of requests for hearing, under proposed 10 CFR 52.99(e)(1). In addition, the staff proposes that the notification be available for public review under proposed 10 CFR 52.99(e)(2). These proposed requirements would ensure public availability and accessibility of all NRC information on ITAAC closure.

Events that affect completed ITAAC could involve activities that include, but are not limited to, maintenance and engineering, program, or design changes. The staff expects that licensees will carry out these activities under established programs to maintain ITAAC conclusions and that no supplemental notification will be necessary in most instances. The NRC can have confidence that prior ITAAC conclusions are maintained as long as the ITAAC determination bases established by the original ITAAC closure letter are not materially altered. If the ITAAC determination bases are not materially altered, licensee activities will remain below the notification threshold of 10 CFR 52.99(c)(3)(i). If the ITAAC determination bases are materially altered, the licensee would be required to notify the NRC under 10 CFR 52.99(c)(3)(i) and

document resolution of the issue in a supplemental ITAAC closure letter under 10 CFR 52.99(c)(3)(ii). In either event, the NRC requires that records documenting the licensee's activities be available at the plant site for NRC review, to comply with proposed 10 CFR 52.99(c)(4), as discussed further in this paper.

All ITAAC Complete Notification

The third new notification that the staff is proposing is the all ITAAC complete notification under 10 CFR 52.99(f)(1). The purpose of this notification is to facilitate the required Commission finding under 10 CFR 52.103(g) that the acceptance criteria in the combined license are met. After or concurrent with the last ITAAC closure letter required by 10 CFR 52.99(c)(1), the licensee would be required to notify the NRC that all ITAAC are complete. At the time the licensee submits the all ITAAC complete notification, the staff would expect that all activities requiring supplemental ITAAC closure letters have been completed and that the associated ITAAC determination bases have been updated.

However, the staff recognizes that construction and operational readiness activities will continue even after the licensee submits the all ITAAC complete notification, and that these activities could result in new information that may materially alter the bases for a finding on acceptance criteria under 10 CFR 52.103(g). The staff understands that prospective licensees expect to complete the last ITAAC very close to the scheduled date for fuel load, and expect the Commission to make the 10 CFR 52.103(g) finding on acceptance criteria shortly thereafter. The NRC's regulatory processes should be structured so that the Commission is able to make a timely 10 CFR 52.103(g) finding that meets all applicable legal standards and is accorded a high level of public confidence. To achieve these objectives, the staff is proposing provisions in 10 CFR 52.99(f)(2) to address a situation where issues occur after submission of the all ITAAC complete notification. This proposed provision requires that if, after filing the all ITAAC complete notification, the licensee identifies new information material to the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met, the licensee must determine whether that information materially alters the basis for the ITAAC determination such that notification would be required under 10 CFR 52.99(c)(3)(i) of this section. If that is the case, the licensee must make the necessary notification within 24 hours of identification of the new information. At this time, the staff would be preparing to make, or may have already made, its recommendation to the Commission in support of the Commission finding under 10 CFR 52.103(g). This prompt notification would be required to ensure that the staff has the most accurate information on which to base its recommendation to the Commission and that the Commission has the most current and accurate information on which to base its finding. The proposed requirement is intended to keep the Commission informed of information material to the finding required under 10 CFR 52.103(g) without being impractical to implement or unduly burdensome.

Note that the staff is proposing to seek specific comment on the time frame for the prompt notification following submittal of the all ITAAC complete notification. The staff believes that 24 hours is a reasonable amount of time for licensees to evaluate whether new information determined to be material to ITAAC closure will, upon further consideration, materially alter the ITAAC determination basis, given the importance of ensuring that the Commission has complete and accurate information at the time it is determining whether the acceptance criteria in the combined license are met. However, because neither the NRC nor the nuclear power

industry have any experience with making these determinations, the staff is interested in feedback on whether its proposal is reasonable.

After making the 24-hour notification, the licensee would be required to submit any supplemental ITAAC closure notification required under 10 CFR 52.99(c)(3)(ii) and to resubmit the all ITAAC complete notification under 10 CFR 52.99(f)(1) to verify that it has no outstanding ITAAC issues. The purpose of this proposed requirement is to ensure that the NRC staff has the most current information on ITAAC closure and that the licensee reaffirms its conclusion that the acceptance criteria for all ITAAC are met before the NRC staff makes (or reiterates) its recommendation to the Commission in support of the 10 CFR 52.103(g) finding.

ITAAC Closure Documentation

In proposed 10 CFR 52.99(c)(4), the staff is adding a provision that would require licensees to maintain records of the bases for determining whether a notification of new information on ITAAC closure under 10 CFR 52.99(c)(3)(i) is required and records of the bases for all notifications required under 10 CFR 52.99(c). Because of the dynamic nature of onsite construction activities and the relatively long period of time from the start of construction until completion, conditions and bases for ITAAC closure could change after notification is made to the NRC under 10 CFR 52.99(c)(1). In addition to supplemental ITAAC closure notifications made under 10 CFR 52.99(c)(3)(ii), the licensee will need to maintain records to reflect changed conditions that do not rise above the threshold that would require a supplemental notification under 10 CFR 52.99(c)(3) (e.g., like-for-like replacement of a component identified in an ITAAC that does not otherwise rise above the reporting thresholds described in the section-by-section analysis for 10 CFR 52.99(c)(3)) (Section III of Enclosure 1). This proposed requirement would ensure that all documentation related to ITAAC closure would be maintained and available for NRC inspection to support the staff's determination about whether the licensee properly performed ITAAC inspections, tests, and analyses and whether the acceptance criteria are met.

The staff proposes to require licensees to retain the records required under proposed 10 CFR 52.99(c)(4) for 5 years after the date the Commission makes the finding under 10 CFR 52.103(g). Although the ITAAC are no longer regulatory requirements after the Commission makes the finding under 10 CFR 52.103(g), a retention period beyond that date is necessary to support enforcement actions related to compliance with ITAAC requirements. The staff is proposing a 5-year retention period because it accords with the statute of limitations for civil penalties found in 28 U.S.C. 2462.

Additional Conforming Changes to 10 CFR 52.99

The staff is also proposing several editorial changes to 10 CFR 52.99(b), (c)(1), (c)(2), and (d)(1). In all of these cases, the staff is proposing to replace the phrase "acceptance criteria have been met" with the phrase "acceptance criteria are met" for consistency with the wording of the requirement in 10 CFR 52.103(g) on the Commission's ITAAC finding, which is derived directly from Section 185.b of the AEA. In addition, the staff is proposing an editorial change to 10 CFR 52.99(d)(2) to replace the phrase "ITAAC has been met" with the phrase "prescribed acceptance criteria are met" for consistency with the wording in 10 CFR 52.99(d)(1).

Conforming Changes to 10 CFR 2.340

The proposed rule contains an associated correction to the language in 10 CFR 2.340(j). The NRC substantially revised 10 CFR 2.340 as part of the 2007 10 CFR Part 52 rulemaking to set forth (among other things) the circumstances under which the Commission may make a finding on acceptance criteria under Section 189.b of the AEA and 10 CFR 52.103(g). The staff's interactions with external stakeholders on ITAAC identified an error in the language of this section. The introductory language of 10 CFR 2.340(j) incorrectly describes the Commission's finding under 10 CFR 52.103(g) as a finding that the acceptance criteria in a combined license "have been, or will be met." This phrase actually describes the finding that a presiding officer makes only with respect to contested matters involving ITAAC in a hearing under 10 CFR 52.103; the presiding officer's finding does not negate the need for the Commission (or appropriate office director, if the Commission chooses to delegate this authority) to make the overall finding on acceptance criteria as required by Section 189.b of the AEA and 10 CFR 52.103(g). Accordingly, the proposed rule contains a correcting change to 10 CFR 2.340(j) to accurately describe the Commission's finding under 10 CFR 52.103(g) as a finding that the acceptance criteria "are met."

In addition to correcting this error, the staff proposes to expand the language of 10 CFR 2.340(j) to (1) more clearly distinguish between adjudicatory findings on contentions (contested matters) with respect to ITAAC and the overall Commission finding on acceptance criteria under Section 189.b of the AEA and 10 CFR 52.103(g), and (2) describe in more detail what findings on ITAAC must be made, depending on whether there are contested ITAAC or not.

COMMITMENT:

Following publication of the proposed rule, the staff will issue draft regulatory guidance for public comment to describe an acceptable method to implement the requirements of this rule.

RECOMMENDATIONS:

The staff recommends that the Commission do the following:

1. Approve for publication in the *Federal Register* the enclosed notice of proposed rulemaking (Enclosure 1).
2. Note the following:
   a. The proposed rule will be published in the *Federal Register* for a 75-day comment period (Enclosure 1).
   b. A draft regulatory analysis has been prepared for this proposed rulemaking (Enclosure 2).
   c. The Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification and the reasons for it, as required by the Regulatory Flexibility Act (5 U.S.C. 605(b)).
   d. The staff has determined that this action is not a "major rule," as defined in the Congressional Review Act of 1996 (5 U.S.C. 804(2)), and has confirmed this

determination with the Office of Management and Budget (OMB). The staff will inform the appropriate congressional and Government Accountability Office contacts.

e. The appropriate congressional committees will be informed.

f. The Office of Public Affairs will issue a press release when the proposed rulemaking is filed with the Office of the Federal Register.

g. OMB review is required. The staff will submit a clearance package to OMB electronically on or immediately after the date the proposed rule is published in the *Federal Register*.

RESOURCES:

Estimated resource needs of 1 full-time equivalent (FTE) position are included in the fiscal year (FY) 2010 budget and 1.1 FTE in the FY 2011 President's Budget as identified below. FY 2012 resources will be addressed through the planning, budget, and performance management process.

OFFICE	FY 2010	FY 2011	FY 2012
Office New Reactors Office	.6 FTE	.8 FTE	.1 FTE
Office of General Counsel	.4 FTE	.1 FTE	
Office Administration		.1 FTE	
Office of Information Systems		.1 FTE	.1 FTE

COORDINATION:

The Office of the General Counsel has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. The Advisory Committee on Reactor Safeguards has deferred its review of this rulemaking until the final rule stage. The rule suggests changes in information collection requirements that must be submitted to OMB on or immediately after the date the proposed rule is published in the *Federal Register*.

/RA by Martin J. Virgilio for/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. *Federal Register* Notice
2. Regulatory Analysis

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 2 and 52

RIN 3150-AI77

NRC-2010-0012

Requirements for Maintenance of Inspections, Tests, Analyses, and Acceptance Criteria

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations related to verification of nuclear power plant construction activities through inspections, tests, analyses, and acceptance criteria (ITAAC) under a combined license. Specifically, the NRC is proposing new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure notification. The new provisions would require licensees to report new information materially altering the basis for determining that a prescribed inspection, test, or analysis was performed as required, or finding that a prescribed acceptance criterion is met; to document the basis for all ITAAC notifications; and to notify the NRC of completion of all ITAAC activities. In addition, the NRC is proposing editorial corrections to existing language in the NRC's regulations to correct and clarify ambiguous language and make it consistent with language in the Atomic Energy Act of 1954, as amended (AEA).

DATES: Submit comments on this proposed rule by [**INSERT DATE 75 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER**]. Submit comments on the information collection aspects on this proposed rule by [**INSERT DATE 30 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER**]. Comments received after the above dates will be considered if it

is practical to do so, but assurance of consideration cannot be given to comments received after these dates.

ADDRESSES: Please include Docket ID **NRC-2010-0012** in the subject line of your comments. You may submit comments by any one of the following methods

Federal Rulemaking Web site: Go to http://www.regulations.gov search for documents filed under Docket ID **NRC-2010-0012.** Address questions about NRC dockets to Carol Gallagher 301-492-3668; e-mail Carol.Gallagher@nrc.gov.

Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

E-mail comments to: Rulemaking.Comments@nrc.gov. If you do not receive a reply e-mail confirming that we have received your comments, contact us directly at 301-415-1677.

Hand deliver comments to: 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. Federal workdays. (Telephone 301-415-1677).

Fax comments to: Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101. You may submit comments on the information collections by the methods indicated in the Paperwork Reduction Act Statement.

See Section VI, Availability of Documents, for instructions on how to access NRC's Agencywide Documents Access and Management System (ADAMS) and other methods for obtaining publicly available documents related to this action.

FOR FURTHER INFORMATION CONTACT: Ms. Nanette V. Gilles, Office of New Reactors, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone at 301-415-1180; e-mail Nanette.Gilles@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Submitting Comments

Comments submitted in writing or in electronic form will be posted on the NRC Web site and on the Federal rulemaking Web site http://www.regulations.gov. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed.

The NRC requests that any party soliciting or aggregating comments received from other persons for submission to the NRC inform those persons that the NRC will not edit their comments to remove any identifying or contact information, and therefore, they should not include any information in their comments that they do not want publicly disclosed.

II. Background

The Commission first issued Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," on April 18, 1989; 54 FR 15371. Section 52.99, "Inspection during construction," was included to make it clear that the NRC's inspection carried out during construction under a combined license would be based on ITAAC proposed by the applicant, approved by the NRC staff, and incorporated in the combined license. At that time, the Commission made it clear that, although 10 CFR 52.99 envisioned a "sign-as-you-go" process in which the NRC staff would sign off on inspection units and notice of the staff's sign-off would be published in the *Federal Register*, the Commission itself would make no findings with respect to construction until construction was complete. See 54 FR 15371; April 18, 1989; at 15383 (second column).

On August 28, 2007 (72 FR 49351), the Commission revised 10 CFR Part 52 to enhance the NRC's regulatory effectiveness and efficiency in implementing its licensing and approval processes. In that revision, the NRC amended 10 CFR 52.99 to require licensees to notify the NRC that the prescribed inspections, tests, and analyses in the ITAAC have been completed and that the acceptance criteria have been met. The revision also requires that these notifications contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. The NRC added this requirement to ensure that combined license applicants and holders were aware that it was the licensee's burden to demonstrate compliance with the ITAAC and the

NRC expected the notification of ITAAC completion to contain more information than just a simple statement that the licensee believes the ITAAC had been completed and the acceptance criteria met.

Under Section 185.b of the Atomic Energy Act of 1954, as amended (AEA) and 10 CFR 52.97(b), a combined license for a nuclear power plant (a "facility") must contain those ITAAC that are "necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with" the license, the AEA, and NRC regulations. Following issuance of the combined license, Section 189.b of the AEA and 10 CFR 52.99(e) require that the Commission "ensure that the prescribed inspections, tests, and analyses are performed." Finally, before operation of the facility, Section 189.b and 10 CFR 52.103(g) require that the Commission find that the "prescribed acceptance criteria *are met*" (emphasis added). This Commission finding will not occur until construction is complete, near the date for scheduled initial fuel load.

As currently required by 10 CFR 52.99(c)(1), the licensee must submit ITAAC closure notifications containing "sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the associated acceptance criteria have been met." These notifications perform two functions. First, they alert the NRC to the licensee's completion of the ITAAC[1] and ensure that the NRC has sufficient information to complete all of the activities necessary for the Commission to determine whether all of the ITAAC acceptance criteria have been or will be met (the "will be met" finding is relevant to any hearing on ITAAC under 10 CFR 52.103) before initial operation. Second, they ensure that interested persons will have access to information on both completed and uncompleted ITAAC at a level of detail

[1] In this discussion, the phrases "completion of ITAAC" and "ITAAC completion" mean that the licensee has determined that: (1) the prescribed inspections, tests, and analyses were performed, and (2) the prescribed acceptance criteria are met.

sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing on acceptance criteria. See 72 FR 49352; August 28, 2007, at 49450 (second column).

After completing the 2007 rulemaking, the NRC began developing guidance on the ITAAC closure process and the requirements under 10 CFR 52.99. In October 2009, the NRC issued regulatory guidance for the implementation of the revised 10 CFR 52.99 in Regulatory Guide (RG) 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52." This RG endorsed guidance developed by the Nuclear Energy Institute (NEI) in NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3, issued January 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090270415).

After considering information presented by industry representatives in a series of public meetings, the NRC realized that some additional implementation issues were left unaddressed by the various provisions in 10 CFR Part 52. In particular, the NRC determined that the combined license holder should provide additional notifications to the NRC following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1). The NRC refers to the time after this ITAAC closure notification, but before the date the Commission makes the finding under 10 CFR 52.103(g), as the ITAAC maintenance period. Most recently, the NRC held two public meetings in March 2010 to discuss draft proposed rule text that it made available to the public in February 2010. The NRC considered feedback given from external stakeholders during those meetings in its development of this proposed rule. Finally, in March 2010, the NRC issued Inspection Procedure 40600, "Licensee Program for ITAAC Management," that provides guidance to verify licensees have implemented ITAAC maintenance programs to ensure that structures, systems, and components continue to meet the ITAAC acceptance criteria until the Commission makes the finding under 10 CFR 52.103(g) allowing operation.

III. Discussion

In brief, the NRC is proposing the following new notifications subsequent to ITAAC closure:

- Notification of new information on ITAAC closure
- Supplemental ITAAC closure notification
- All ITAAC complete notification

In general, the reasons for these proposed new notifications are analogous to the reasons presented in the 2007 rulemaking for the existing 10 CFR 52.99(c) notifications (i.e., to ensure that the NRC has sufficient information, in light of new information developed or identified after the ITAAC closure notification under 10 CFR 52.99(c)(1) to complete all of the activities necessary for the Commission to make a determination on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing). Each of the proposed notification and documentation requirements in this rulemaking, and the basis for each of the proposed requirements, are described in Section III.B, "Additional ITAAC Notifications and Documentation," of this document.

The NRC is also proposing several editorial changes to 10 CFR 52.99 in paragraphs (b), (c)(1), (c)(2), and (d)(1). In all of these cases, the NRC is proposing to replace the phrase "acceptance criteria have been met" with the phrase "acceptance criteria are met" for consistency with the wording of the requirement in 10 CFR 52.103(g) on the Commission's ITAAC finding, which is derived directly from wording in the AEA.

In addition, the NRC is proposing an editorial change to 10 CFR 52.99(d)(2) to replace the phrase "ITAAC has been met" with the phrase "prescribed acceptance criteria are met" for consistency with the wording in 10 CFR 52.99(d)(1).

A. Licensee Programs That Maintain ITAAC Conclusions

One essential element in ensuring the maintenance of successfully completed ITAAC involves the use of established licensee programs such as the Quality Assurance Program, Problem Identification and Resolution Program, Maintenance/Construction Program, and Design and Configuration Management Program. Each program credited with supporting the maintenance of completed ITAAC should contain attributes that maintain the validity of the ITAAC determination basis. These program attributes include the following:

- Licensee screening of activities and events for impact on ITAAC;
- Licensee determination of whether supplemental ITAAC notification is required; and
- Licensee supplement of the ITAAC closure package as appropriate to demonstrate that the acceptance criteria continue to be met.

The NRC expects these programs to be fully implemented and effective before the licensee takes credit for them as an appropriate means of supporting ITAAC maintenance. These programs will be subject to NRC inspection.

B. Additional ITAAC Notifications and Documentation

The NRC's confidence in the licensee's ability to maintain the validity of completed ITAAC conclusions relies on timely communication. Currently, 10 CFR 52.99 specifies two ITAAC notification requirements for licensees. These notifications are the ITAAC closure notification required by 10 CFR 52.99(c)(1) and the notification of uncompleted ITAAC required by 10 CFR 52.99(c)(2) no less than 225 days before scheduled fuel load. The NRC believes that additional formal notifications to the NRC are needed that are not currently required by regulation.

Notification of New Information on ITAAC Closure

The first new notification is contained in proposed 10 CFR 52.99(c)(3)(i), "New information on ITAAC closure," and would require licensees to notify the NRC of new

information materially altering the basis for determining that an inspection, test, or analysis was performed as required, or finding that an acceptance criterion is met (referred to as the *ITAAC determination basis*). The notification must be made no later than 7 days after the licensee has determined that the new information materially alters the ITAAC determination basis.

The purpose of this initial notification would be to alert the NRC staff responsible for oversight of construction inspection activities to the fact that additional activities may be scheduled that affect a structure, system, or component (including physical security hardware) or program element for which one or more ITAAC have been closed. This will allow the NRC inspection staff to discuss the licensee plans for resolving the issue to determine if the staff wants to observe any of the upcoming activities for the purpose of making a future staff determination about whether the acceptance criteria for those ITAAC continue to be met.

The licensee is responsible for maintaining the validity of the ITAAC conclusions after completion of the ITAAC. If the ITAAC determination basis is materially altered, the licensee is expected to notify the NRC. Through public workshops and stakeholder interaction, the NRC has developed thresholds to identify when activities would materially alter the basis for determining that a prescribed inspection, test, or analysis was performed as required, or finding that a prescribed acceptance criterion is met. One obvious case is that a notification under proposed paragraph (c)(3)(i) would be required to correct a material error or omission in the original ITAAC closure notification.

Section 52.6, "Completeness and accuracy of information," paragraph (a), requires that information provided to the Commission by a licensee be complete and accurate in all material respects. However, it might be the case that the original closure notification was complete and accurate when sent, but subsequent events materially alter the ITAAC determination bases. Also, a material error or omission might not be discovered until after the ITAAC closure notification is sent. It is possible that new information materially altering the ITAAC

determination bases would not rise to the reporting threshold under 10 CFR 52.6(b). As required by 10 CFR 52.6(b), licensees must notify the Commission of information identified by the licensee as having, for the regulated activity, a significant implication for public health and safety or the common defense and security. Given the primary purpose of ITAAC—to verify that the plant has been constructed and will be operated in compliance with the approved design—the NRC believes that it cannot rely on the provisions in 10 CFR 52.6 for licensee reporting of new information materially altering the ITAAC determination bases. The reasons for this conclusion are as follows:

1. Material errors and omissions in ITAAC closure notifications, relevant to the accuracy and completeness of the documented bases for the Commission's finding on ITAAC, may nonetheless be determined in isolation by a licensee as not having a significant implication for public health and safety or common defense and security.
2. A Commission finding of compliance with acceptance criteria in the ITAAC is required, under Section 185.b of the AEA, in order for the combined license holder to commence operation.
3. The addition of specific reporting requirements addressing information relevant and material to the ITAAC finding ensures that NRC will get the necessary reports as a matter of regulatory requirement, and allows the NRC to determine the timing and content of these reports so that they serve the regulatory needs of the NRC.

Therefore, the NRC intends that these issues will be reported under 10 CFR 52.99(c)(3)(i). In addition to the reporting of material errors and omissions, the NRC has identified other circumstances in which reporting under this provision would be required (i.e., reporting thresholds). These reporting thresholds are described in more detail in the Section IV,

"Section-By-Section Analysis," of this document.

Supplemental ITAAC Closure Notification

After receiving the new information notification under proposed 10 CFR 52.99(c)(3)(i), the NRC's interest in the circumstances requiring that notification would not end. When making the 10 CFR 52.103(g) finding, the NRC must have information sufficient to determine that the relevant acceptance criteria are met despite the new information provided in the notification under paragraph (c)(3)(i). The licensee's summary statement of the basis for resolving the issue, which was the subject of the new information notification, a discussion of any action taken, and a list of the key licensee documents supporting the resolution and its implementation, would assist the NRC in making its independent evaluation of the issue. Apart from the NRC's use of the information, the NRC also believes that public availability of such information is necessary to ensure that interested persons will have sufficient information to review when preparing a request for a hearing under 10 CFR 52.103, comparable to the information provided under paragraph (c)(1), as described in the statement of consideration for the 2007 Part 52 rulemaking. See August 28, 2007; 72 FR 49352, at 49384 (second and third column). Accordingly, the NRC proposes that after a licensee makes a new information notification under paragraph (c)(3)(i), it must then submit what is essentially a "resolution" notification to the NRC in the form of a supplemental ITAAC closure notification. This supplemental ITAAC closure notification, described in proposed paragraph (c)(3)(ii), would require the licensee to submit a written notification of the resolution of the circumstances that prompted the paragraph (c)(3)(i) new information notification. The supplemental ITAAC closure notification must contain sufficient information demonstrating that, notwithstanding the information that prompted the paragraph (c)(3)(i) notification, the prescribed inspections, tests and analyses have been performed as required and the prescribed acceptance criteria are met. Supplemental ITAAC closure notifications should explain the need for the supplemental notification, outline the

resolution of the issue, and confirm that the ITAAC acceptance criteria continue to be met. Supplemental ITAAC closure notifications must include a level of detail similar to the level of information required in initial ITAAC closure notifications under 10 CFR 52.99(c)(1).

The NRC proposes that the 10 CFR 52.99(c)(3)(ii) notification be made within 30 days after licensee resolution of the issue. The NRC selected the 30-day period as a reasonable compromise between the competing goals of prompt notification and the recognition that documentation of the resolution may be quite extensive, depending on the significance and number of activities necessary for the licensee to resolve the issue. The NRC proposes that NRC receipt of the notification be published in the *Federal Register*, at least until the last date for submission of requests for hearing, under proposed paragraph (e)(1). In addition, the NRC proposes that the notification be available for public review under proposed paragraph (e)(2). These two proposed requirements would ensure public availability and accessibility of all NRC information on ITAAC closure. Further explanation of the basis for the publication and availability requirements is presented further under the discussion on proposed 10 CFR 52.99(e)(1) and (e)(2).

Events that affect completed ITAAC could involve activities that include, but are not limited to, maintenance and engineering, program, or design changes. The NRC expects that licensees will carry out these activities under established programs to maintain ITAAC conclusions and that no supplemental notification will be necessary in most instances. The NRC can have confidence that prior ITAAC conclusions are maintained as long as the ITAAC determination basis established by the original ITAAC closure notification is not materially altered. If the ITAAC determination basis is not materially altered, licensee activities will remain below the notification threshold of 10 CFR 52.99(c)(3)(i). If the ITAAC determination basis is materially altered, the licensee would be required to notify the NRC under 10 CFR 52.99(c)(3)(i) and document resolution of the issue in a supplemental ITAAC closure notification under

10 CFR 52.99(c)(3)(ii). In either event, records documenting the licensee's activities are required to be available at the plant site for NRC review to comply with 10 CFR 52.99(c)(4), as discussed further in this document.

ITAAC Closure Documentation

In proposed 10 CFR 52.99(c)(4), the NRC is adding a provision that would require that licensees maintain records of the basis for determining whether a notification of new information on ITAAC closure under 10 CFR 52.99(c)(3)(i) is required and records of the bases for all notifications required under 10 CFR 52.99(c). Because of the dynamic nature of onsite construction activities, and the relatively long period of time from the start of construction until construction completion, conditions and bases for ITAAC closure could change after notification is made to the NRC under 10 CFR 52.99(c)(1). In addition to supplemental ITAAC closure notifications made under paragraph (c)(3)(ii), the licensee will need to maintain records to reflect changed conditions that do not rise above the threshold that would require a supplemental notification under paragraph (c)(3) (e.g., like-for-like replacement of a component identified in an ITAAC that does not otherwise rise above the reporting thresholds described in the section-by-section analysis in this document for 10 CFR 52.99(c)(3)). This proposed requirement would ensure that all documentation related to ITAAC closure would be maintained and available for NRC inspection to support the NRC staff's determination about whether the licensee properly performed ITAAC inspections, tests, and analyses and whether the acceptance criteria are met.

The NRC is proposing to require that licensees retain the records required under proposed paragraph (c)(4) for 5 years after the date the Commission makes the finding under 10 CFR 52.103(g). The NRC believes that, although the ITAAC are no longer regulatory requirements after the Commission makes the finding under 10 CFR 52.103(g), a retention period beyond that date is necessary to support enforcement actions related to compliance with

ITAAC requirements. The NRC is proposing a 5-year retention period because it accords with the statute of limitations for civil penalties found in 28 U.S.C. 2462.

NRC inspection, publication of notices, and availability of licensee notifications

Section 52.99(e)(1) requires that NRC publish in the *Federal Register* the NRC staff's determination of the successful completion of inspections, tests, and analyses, at appropriate intervals until the last date for submission of requests for hearing under 10 CFR 52.103(a). The NRC is proposing revisions to paragraph (e)(1) to include notices in the *Federal Register* of the licensee's notification that the determination basis for a completed ITAAC has been materially altered under paragraph (c)(3)(i) and any NRC staff determination that the acceptance criteria for the affected ITAAC are met following a licensee notification under paragraph (c)(3)(ii) of this section.

Section 52.99(e)(2) currently provides that the NRC shall make publicly available the licensee notifications under paragraphs (c)(1) and (c)(2). The NRC is proposing to revise paragraph (e)(2) to cover all notifications under 10 CFR 52.99(c) and (f). In general, the NRC expects to make the paragraphs (c) and (f) notifications available shortly after the NRC has received the notifications and concluded that they are complete. Furthermore, by the date of the *Federal Register* notice of intended operation and opportunity to request a hearing on whether acceptance criteria are met (under 10 CFR 52.103(a)), the NRC will make available the licensee notifications under paragraphs (c)(1), (c)(2), and (c)(3) that it has received to date.

All ITAAC Complete Notification

Another notification that the NRC is proposing is the all ITAAC complete notification under 10 CFR 52.99(f)(1). The purpose of this notification is to facilitate the required Commission finding under 10 CFR 52.103(g) that the acceptance criteria in the combined license are met. After or concurrent with the last ITAAC closure notification required by

10 CFR 52.99(c)(1), the licensee would be required to notify the NRC that all ITAAC are complete. When the licensee submits the all ITAAC complete notification, the NRC would expect that all activities requiring supplemental ITAAC closure notifications have been completed and that the associated ITAAC determination bases have been updated.

However, the NRC recognizes that construction and operational readiness activities will continue even after the licensee submits the all ITAAC complete notification, and that these activities could result in new information that may materially alter the basis for a finding on acceptance criteria under 10 CFR 52.103(g). The NRC understands that prospective licensees expect to complete the last ITAAC very close to the scheduled date for fuel load, and expect the Commission to make the 10 CFR 52.103(g) finding on acceptance criteria shortly thereafter. The NRC's regulatory processes should be structured so that the Commission is able to make a timely 10 CFR 52.103(g) finding that meets all applicable legal standards and is accorded a high level of public confidence. To achieve these objectives, the NRC is proposing provisions in 10 CFR 52.99(f)(2) to address a situation where issues occur after submission of the all ITAAC complete notification. This proposed provision states that if, after filing the all ITAAC complete notification, the licensee identifies new information material to the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met, the licensee shall determine whether notification is required under paragraph (c)(3)(i) of this section and make the necessary notification within 24 hours of identification of the new information. At this time, the NRC staff would be preparing to make, or may have already made, its recommendation to the Commission in support of the Commission finding under 10 CFR 52.103(g). This prompt notification would be required to ensure that the NRC staff has the most accurate information on which to base its recommendation to the Commission and that the Commission has the most current and accurate information on which to base its finding. The proposed requirement is intended to keep the Commission informed of

information material to the finding required under 10 CFR 52.103(g) without being impractical to implement or unduly burdensome.

After making the 24-hour notification, the licensee would be required to submit any further notification required under 10 CFR 52.99(c)(3)(ii) and to resubmit the all ITAAC complete notification under 10 CFR 52.99(f)(1) to verify that it has no outstanding ITAAC issues. The purpose of this proposed requirement is to ensure that the NRC staff has the most current information on ITAAC closure and that the licensee reaffirms its conclusion that the acceptance criteria for all ITAAC are met before the NRC staff makes (or reiterates) its recommendation to the Commission in support of the 10 CFR 52.103(g) finding.

As stated previously, the purpose of the all ITAAC complete notification is to facilitate the required Commission finding under 10 CFR 52.103(g) that the acceptance criteria in the combined license are met. To support the Commission's finding, if and when appropriate, the NRC staff will send a recommendation to the Commission. The staff will consider that all ITAAC "are met" if both of the following conditions hold:

- All ITAAC were verified to be met at one time; and
- The licensee provides confidence that the ITAAC determination bases have been maintained and that the ITAAC acceptance criteria continue to be met.

The staff approach would allow licensees to have ITAAC-related structures, systems, or components, or security or emergency preparedness related hardware, undergoing certain activities at the time of the 10 CFR 52.103(g) finding, if the programs credited with maintaining the validity of completed ITAAC, guide those activities and the activities are not so significant as to exceed a threshold for reporting. If a reporting threshold has been exceeded, the NRC would need to evaluate the licensee's supplemental ITAAC closure notification to determine whether the ITAAC continue to be met. Reporting thresholds are discussed in more detail in the section-by-section analysis for 10 CFR 52.99(c)(3) in this document.

C. Conforming Changes to 10 CFR 2.340

The 2007 Part 52 rulemaking amended 10 CFR 2.340, "Initial decision in certain contested proceedings; immediate effectiveness of initial decisions; issuance of authorizations, permits, and licenses," to clarify, among other things, the scope of the presiding officer's decision in various kinds of NRC proceedings, and remove the requirement for direct Commission involvement in all production and utilization facility licensing proceedings.

Section 2.340(j) was intended to address these matters in connection with the Commission finding on acceptance criteria and any associated hearing under 10 CFR 52.103. In the course of developing this proposed rule, the NRC determined that 10 CFR 2.340(j) contains several errors and ambiguous statements. The proposed changes, together with the proposed bases for the changes, are described below.

Section 2.340(j) currently states that the Commission makes a finding under 10 CFR 52.103(g) that acceptance criteria "have been or will be met." This is incorrect; the Commission's finding under 10 CFR 52.103(g) is that the acceptance criteria "are met," which is the statutory requirement under Section 185.b of the AEA. To correct this error, the NRC proposes to amend the introductory language of 10 CFR 2.340(j) to use the correct phrase, "acceptance criteria...are met...."

In addition, 10 CFR 2.340(j), as currently written, does not clearly address the circumstances in a contested proceeding that could lead to a Commission finding under 10 CFR 52.103(g) that acceptance criteria are met. To provide clarity, the NRC proposes to further amend 10 CFR 2.340(j) to clearly explain when the Commission may make the 10 CFR 52.103(g) finding, by further delineating between the presiding officer's decisions on contentions that acceptance criteria have not been met and decisions on contentions that acceptance criteria will not be met. In both cases, if the presiding officer's decision resolves the contention favorably this does not obviate the need for the Commission to make the required

finding under Section 185.b of the AEA and 10 CFR 52.103(g) that the acceptance criteria are met. For example, the presiding officer's initial decision upon summary disposition that a particular acceptance criterion has been met may be rendered before the occurrence of an event which is ultimately resolved as reported in a 10 CFR 52.99(c)(3)(ii) notification. In such a circumstance, the Commission must independently come to the conclusion that the acceptance criterion is met. That conclusion must be based upon consideration of both the presiding officer's initial decision and information relevant to the 10 CFR 52.99(c)(3)(ii) notification. Accordingly, the NRC concludes that it is necessary to clarify the language of paragraph (j). To accommodate the proposed clarifications, the Commission proposes to redesignate current paragraph (2) as paragraph (4), but without any change to the regulatory language.

IV. Section-by-Section Analysis

The primary changes on ITAAC maintenance being proposed by the NRC in this rulemaking are to 10 CFR 52.99. The changes to 10 CFR 2.340 are corrections.

Section 2.340 Initial decision in certain contested proceedings; immediate effectiveness of initial decisions; issuance of authorizations, permits and licenses.

Section 2.340(j) Issuance of finding on acceptance criteria under 10 CFR 52.103.

Paragraph (j) would be amended to allow the Commission (or the appropriate staff Office Director) in a contested proceeding to make the finding under 10 CFR 52.103(g) that the acceptance criteria in a combined license are met, under certain circumstances that are delineated in greater detail in paragraphs (j)(1) through (4). This compares with the current rule, which contains only two paragraphs (j)(1) and (2). The matters covered by paragraph (j)(1) of the current rule would be described with greater clarity in proposed paragraphs (j)(1) through (3).

Proposed paragraph (j)(1) clarifies that the Commission may not make the overall 10 CFR 52.103(g) finding unless it is otherwise able to find that all uncontested acceptance

criteria (i.e., "acceptance criteria not within the scope of the initial decision of the presiding officer") are met. The phrase "otherwise able to make" conveys the NRC's determination that the Commission's process for supporting a Commission finding on uncontested acceptance criteria is unrelated to and unaffected by the timing of the presiding officer's initial decision.

Proposed paragraph (j)(2) clarifies that a presiding officer's initial decision which finds that acceptance criteria have been met, is a necessary but not sufficient prerequisite for the Commission to make a finding that the contested acceptance criteria (i.e., the criteria which are the subject of the presiding officer's initial decision) are met. The Commission must thereafter – even if the presiding officer's initial decision finds that the contested acceptance criteria have been met – be able to make a finding that the contested criteria are met after considering: (1) information submitted in the licensee notifications which the NRC proposes to be included in 10 CFR 52.99; and (2) the NRC staff's findings with respect to these notifications, to issue the overall 10 CFR 52.103 finding. By using the word "thereafter," the NRC intends to emphasize that the Commission would not make a finding that contested acceptance criteria are met in advance of the presiding officer's initial decision on those acceptance criteria.

Proposed paragraph (j)(3) expresses the same concept as paragraph (j)(2) but as applied to findings that acceptance criteria will be met. Thus, even if a presiding officer's initial decision finds that the contested acceptance criteria will be met, the Commission must thereafter be able to make a finding that the contested criteria are met after considering: (1) information submitted in an ITAAC closure notification pursuant to 10 CFR 52.99(c)(1); (2) information submitted in the licensee notifications which the NRC proposes to be included in 10 CFR 52.99; and (2) the NRC staff's findings with respect to such notifications, to issue the overall 10 CFR 52.103 finding.

Proposed paragraph (j)(4) is the same as the existing provision in 10 CFR 2.340(j)(2). This paragraph provides that the Commission may make the 52.103(g) finding notwithstanding

the pendency of a petition for reconsideration under 10 CFR 2.345, a petition for review under 10 CFR 2.341, a motion for a stay under 10 CFR 2.342, or a petition under 10 CFR 2.206.

The NRC notes that 10 CFR 2.340(j) is not intended to be an exhaustive "roadmap" to a possible 10 CFR 52.103(g) finding that acceptance criteria are met. For example, this provision does not directly address what must occur for the Commission to make a 10 CFR 52.103(g) finding where the presiding officer finds, with respect to a contention, that acceptance criteria are *not* met. The NRC also notes that this provision applies only to contested proceedings. If there is no hearing under 10 CFR 52.103, or if the hearing ends without a presiding officer's initial decision on the merits (e.g., a withdrawal of the sole party in a proceeding), then 10 CFR 2.340(j) does not govern the process by which the Commission (or the appropriate staff Office Director) makes the 10 CFR 52.103(g) finding.

Section 52.99 Inspection during construction; ITAAC schedules and notifications; NRC notices.

Although the NRC is not making changes to every paragraph under 10 CFR 52.99, for simplicity, this rulemaking would replace the section in its entirety. Therefore, the NRC is providing a section-by-section discussion for every paragraph in 10 CFR 52.99. For those paragraphs where little or no change is being proposed, the NRC is repeating the section-by-section discussion from the 2007 major revision to 10 CFR Part 52 (72 FR 49450-49451; August 28, 2007) with editorial and conforming changes, as appropriate.

The purpose of this section is to present the requirements to support the NRC's inspections during construction, including requirements for ITAAC schedules and notifications and for NRC notices of ITAAC closure.

Section 52.99(a) Licensee schedule for completing inspections, tests or analyses.

The NRC is not proposing any changes to this paragraph. Paragraph (a) requires that the licensee submit to the NRC, no later than 1 year after issuance of the combined license or at the start of construction as defined at 10 CFR 50.10, whichever is later, its schedule for

completing the inspections, tests, or analyses in the ITAAC. This provision also requires the licensee to submit updates to the ITAAC schedule every 6 months thereafter and, within 1 year of its scheduled date for initial loading of fuel, licensees must submit updates to the ITAAC schedule every 30 days until the final notification is provided to the NRC under 10 CFR 52.99(c). The information provided by the licensee will be used by the NRC in developing the NRC's inspection activities and activities necessary to support the Commission's finding whether all of the ITAAC are met prior to the licensee's scheduled date for fuel load. Even in the case where there were no changes to a licensee's ITAAC schedule during an update cycle, the NRC expects the licensee to notify the NRC that there have been no changes to the schedule.

Section 52.99(b) Licensee and applicant conduct of activities subject to ITAAC.

The NRC is proposing an editorial change to the last sentence of 10 CFR 52.99(b) to replace the words "have been met" with "are met" for consistency with the requirements of Section 185.b of the AEA, as implemented in 10 CFR 52.103(g). The purpose of the requirement in 10 CFR 52.99(b) is to clarify that an applicant may proceed at its own risk with design and procurement activities subject to ITAAC, and that a licensee may proceed at its own risk with design, procurement, construction, and preoperational testing activities subject to an ITAAC, even though the NRC may not have found that any particular ITAAC are met.

Section 52.99(c) Licensee notifications and documentation.

Section 52.99(c)(1) ITAAC closure notification and Section 52.99(c)(2) Uncompleted ITAAC notification.

The NRC is proposing editorial changes in 10 CFR 52.99(c)(1) to replace the words "have been met" with "are met." Section 52.99(c)(1) would require the licensee to notify the NRC that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met. Section 52.99(c)(1) would further require that the

notification contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met.

The NRC is proposing an editorial change to the last sentence in 10 CFR 52.99(c)(2) to replace the words "have been met" with "are met." Paragraph 52.99(c)(2) requires that, if the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by paragraph (c)(1) of this section for all ITAAC, then the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation (consistent with the AEA Section 185.b requirement that the Commission, "prior to operation," find that the acceptance criteria in the combined license are met). The notification must be provided no later than the date 225 days before the scheduled date for initial loading of fuel, and must provide sufficient information to demonstrate that the prescribed inspections, tests, or analyses will be performed and the prescribed acceptance criteria for the uncompleted ITAAC will be met.

Section 52.99(c) ensures that: (1) the NRC has sufficient information to complete all of the activities necessary for the Commission to make a finding as to whether all of the ITAAC are met prior to initial operation; and (2) interested persons will have access to information on both completed and uncompleted ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing on acceptance criteria. It is the licensee's burden to demonstrate compliance with the ITAAC and the NRC expects the information submitted under paragraph (c)(1) to contain more than just a simple statement that the licensee believes the ITAAC has been completed and the acceptance criteria met. The NRC would expect the notification to be sufficiently complete and detailed for a reasonable person to understand the bases for the licensee's representation that the inspections, tests, and analyses have been successfully completed and the acceptance criteria are met. The term "sufficient information"

would require, at a minimum, a summary description of the bases for the licensee's conclusion that the inspections, tests, or analyses have been performed and that the prescribed acceptance criteria are met.

Furthermore, with respect to uncompleted ITAAC, it is the licensee's burden to demonstrate that it will comply with the ITAAC and the NRC would expect the information that the licensee submits under paragraph (c)(2) to be sufficiently detailed such that the NRC staff can determine what activities it will need to undertake to determine if the acceptance criteria for each of the uncompleted ITAAC are met, once the licensee notifies the NRC that those ITAAC have been successfully completed and their acceptance criteria met. The term "sufficient information" requires, at a minimum, a summary description of the bases for the licensee's conclusion that the inspections, tests, or analyses will be performed and that the prescribed acceptance criteria will be met. In addition, "sufficient information" includes, but is not limited to, a description of the specific procedures and analytical methods to be used for performing the inspections, tests, and analyses and determining that the acceptance criteria are met.

The NRC notes that, even though it did not include a provision requiring the completion of all ITAAC by a certain time prior to the licensee's scheduled fuel load date, the NRC staff will require some period of time to perform its review of the last ITAAC once the licensee submits its notification that the ITAAC has been successfully completed and the acceptance criteria met. In addition, the Commission itself will require some period of time to perform its review of the staff's conclusions regarding all of the ITAAC and the staff's recommendations regarding the Commission finding under 10 CFR 52.103(g).

Section 52.99(c)(3) ITAAC post-closure notifications.

The NRC is proposing to add new paragraph (c)(3) that would require two notifications subsequent to ITAAC closure. The first is an early notification of new information on ITAAC closure under paragraph (c)(3)(i). The second proposed notification under paragraph (c)(3)(ii) is

the supplemental ITAAC closure notification documenting the resolution of issues identified in the earlier notification under paragraph (c)(3)(i).

Section 52.99(c)(3)(i) New information on ITAAC closure.

Paragraph (c)(3)(i) would require licensees to notify the NRC of new information materially altering the basis for determining that an inspection, test or analysis was performed as required, or that an acceptance criterion is met. The NRC is proposing that the notification be made no later than 7 days after licensee determination that the information materially alters the ITAAC determination basis.

Fundamentally, those circumstances requiring notification under paragraph (c)(3)(i) fall into the following two categories:

- The information presented or referenced in the original 10 CFR 52.99(c)(1) notification is insufficient, either because it omits material information, or because the information is materially erroneous or incorrect, and the licensee discovers or determines there is a material omission or error after filing the original 10 CFR 52.99(c)(1) notification.
- The information presented or referenced in the original 10 CFR 52.99(c)(1) notification was complete (i.e., not omitting material information) and accurate (i.e., not materially erroneous), but there is new material information with respect to the subject of the original 10 CFR 52.99(c)(1) notification.

The term "materially altering" refers to situations in which there is information not contained in the 10 CFR 52.99(c)(1) notification that "has a natural tendency or capability to influence an agency decision maker" in either determining whether the prescribed inspection, test or analysis was performed as required, or finding that the prescribed acceptance criterion is met. See Final Rule; Completeness and Accuracy of Information, December 31, 1987; 52 FR 49362, at 49363. Applying this concept in the context of 10 CFR 52.99(c), information for which notification would be required under paragraph (c)(3)(i) is that information which,

considered by itself or when considered in connection with information previously submitted or referenced by the licensee in a paragraph (c)(1) notification, relates to information which is necessary for any of the following:

- The licensee to assert that the prescribed inspections, test, and analyses have been performed and the acceptance criteria are met;
- The NRC staff to determine if (and provide a recommendation to the Commission as to whether) the prescribed inspections, tests, and analyses were performed and the acceptance criteria are met; or
- The Commission to find that the acceptance criteria are met, as required by Section 185.b of the AEA and 10 CFR 52.103(g).

The term "new" information embraces three different kinds of information:

- New information (i.e., a "discovery" or new determination identified after the 10 CFR 52.99(c)(1) notification) about the accuracy of material information provided in, referenced by, or necessary to support representations made in that notification.
- New information (i.e., a "discovery" or new determination identified after the 10 CFR 52.99(c)(1) notification) that previously existing information should have been, but was not provided in the notification or referenced in the supporting documentation (i.e., an omission of material information).
- Information on a "new" event or circumstance (i.e., an event or circumstance occurring after the 10 CFR 52.99(c)(1) notification) that materially affects the accuracy or completeness of the basis – as reported or relied upon in the 52.99(c)(1) notification – for the licensee's representation that the acceptance criteria are met.

Applying these concepts, the NRC believes that the circumstances for which reporting under this provision would be required include:

- *Material Error or Omission* – Is there a material error or omission in the original ITAAC closure notification?
- *Post Work Verification (PWV)* - Will the PWV performed following work undertaken to resolve an issue reportable under 10 CFR 52.99(c)(3)(i) use a significantly different approach than the original performance of the inspection, test, or analysis as described in the original ITAAC notification?
- *Engineering Changes* - Will an engineering change be made that materially alters the determination that the acceptance criteria are met?
- *Additional Items to be Verified* - Will there be additional items that need to be verified through the ITAAC?
- *Complete and Valid ITAAC Representation* - Will any other licensee activities materially alter the ITAAC determination basis?

Additional guidance on implementing these reporting thresholds is being developed by NEI as a revision to its existing ITAAC closure guidance in NEI 08-01. The NRC staff will issue a proposed revision to RG 1.215 for public comment and is considering endorsing the revised guidance prepared by NEI. This proposed guidance is discussed further in Section V, "Guidance," of this document.

The NRC is proposing that notifications under 10 CFR 52.99(c)(3)(i) be made to the NRC Operations Center. The NRC's preferred notification method is by e-mail to hoo.hoc@nrc.gov. Notification can also be made by facsimile to the NRC Operations Center at (301) 816–5151, or by telephone at (301) 816–5100. Verification that the facsimile has been received should be made by calling the NRC Operations Center.

Notifications made to the NRC Operations Center under 10 CFR 52.99(c)(3)(i) should clearly indicate that they are being submitted under 10 CFR 52.99 (c)(3)(i) and include, but need not be limited to, the following information, to the extent known:

- Name , address, telephone number, and title or position within licensee organization of individual or individuals informing the Commission.
- Identification of the facility reporting the situation that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- The date that the licensee determined that it had information that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- The specific ITAAC affected and the date of the original ITAAC closure notification submitted under 10 CFR 52.99(c)(1).
- The systems affected and the nature of the condition that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- Whether any corrective actions are planned during the next 24 hours.
- Whether the onsite NRC Resident Inspector has been informed (although the report should include information about whether the Resident Inspector has been informed, this is not a prerequisite for making the report and notification of the Resident Inspector should not inhibit making a timely notification to the NRC Operations Center).

Section 52.99(c)(3)(ii) Supplemental ITAAC closure notification.

Proposed paragraph (c)(3)(ii) would require the licensee, after submitting a notification under paragraph (c)(3)(i), to submit a supplemental ITAAC closure notification documenting the resolution of the issue which prompted the paragraph (c)(3)(i) report. By "resolution," the NRC means: (1) the completion of the licensee's technical evaluation of the issue and the determination as to whether the prescribed inspection, test, or analysis was performed as required; (2) licensee completion of any necessary corrective or supplemental actions; (3)

licensee documentation of the issue and any necessary corrective or supplemental actions in order to bring the ITAAC determination basis up to date; and (4) ultimate licensee determination about whether the affected acceptance criteria continue to be met.

The information provided in the notification should be at a level of detail comparable to the ITAAC closure notification under paragraph (c)(1). The dual purposes of the proposed paragraph (c)(3)(ii) notification, as described in Section III.B, "Additional ITAAC Notifications and Documentation," of this document, are comparable to the purposes of the ITAAC closure notification in paragraph (c)(1). Thus, the NRC believes that the considerations for the content of the ITAAC closure notification, as discussed in the final 2007 Part 52 rule, apply to the proposed paragraph (c)(3)(ii) notifications. See 72 FR 49450; August 28, 2007 (second column). Thus, it is the licensee's burden to demonstrate compliance with the ITAAC, taking into account any new information that materially alters the determination that a prescribed inspection, test or analysis was performed as required or that a prescribed acceptance criterion is met. The NRC expects the paragraph (c)(3)(ii) notification to contain more than just a simple statement that the licensee has concluded, despite the new information which prompted the paragraph (c)(3)(i) notification, that the prescribed inspection, test or analysis was performed as required and that a prescribed acceptance criterion is met. The NRC expects the notification to be sufficiently complete and detailed for a reasonable person to understand the bases for the licensee's determination in the paragraph (c)(3)(ii) notification. The term "sufficient information" is comparable to the meaning given to that term in paragraph (c)(1), and requires, at a minimum, a summary description of the bases for the licensee's determination. In addition, "sufficient information" includes, but is not limited to, a description of the specific procedures and analytical methods used or relied upon to develop or support the licensee's determination. The notification must be submitted to the NRC within 30 days of resolution of the issue which prompted the paragraph (c)(3)(i) notification. The paragraph (c)(3)(ii) notification must be in

writing, and the records on which it is based must be retained by the licensee to comply with the requirements of proposed 10 CFR 52.99(c)(4). Licensees should use the same process for submitting supplemental ITAAC closure notifications as would be used to submit initial ITAAC closure notifications. The NRC is issuing draft guidance on implementation of the requirements in proposed paragraph (c)(3), including the level of detail necessary to comply with the requirements of proposed paragraph (c)(3)(ii), as discussed in Section V of this document.

Section 52.99(c)(4) ITAAC closure documentation.

Paragraph 10 CFR 52.99(c)(4) would require that licensees maintain records of the bases for determining whether a notification of new information on ITAAC closure under 10 CFR 52.99(c)(3)(i) is required and records of the bases for all notifications required under 10 CFR 52.99(c). This would include records supporting initial ITAAC closure notifications under paragraph (c)(1), uncomplete ITAAC notifications under paragraph (c)(2), and supplemental ITAAC closure notifications under paragraph (c)(3). The onsite ITAAC closure package would provide the technical basis for the licensee's submittals under 10 CFR 52.99(c). As such, it can be viewed as a "roadmap" documenting how the licensee has established that the activities related to verifying that the ITAAC acceptance criteria are met were accomplished. Documents reviewed and referenced in the ITAAC closure or supplemental closure notifications and key supporting documents should be listed in the closure package and should be readily retrievable for ease of verification by the NRC during inspections. If certain supporting information is not available onsite, the ITAAC closure package should indicate where that information may be inspected or audited, if necessary. Licensees would be required to retain these records for a period of 5 years after the date the Commission makes the finding under 10 CFR 52.103(g).

Section 52.99(d) Licensee determination of non-compliance with ITAAC.

Paragraph (d) states the options that a licensee will have in the event that it is determined that any of the acceptance criteria in the ITAAC are not met. If an activity is subject to an ITAAC derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC are met, the licensee may take corrective actions to successfully complete that ITAAC or request an exemption from the standard design certification ITAAC, as applicable. A request for an exemption must also be accompanied by an application for a license amendment under 10 CFR 52.98(f). The NRC will consider and take action on the request for exemption and the license amendment application together as an integrated NRC action.

Also, if an activity that is subject to an ITAAC not derived from a referenced standard design certification and the licensee has not demonstrated that the ITAAC has been met, the licensee may take corrective actions to successfully complete that ITAAC or request a license amendment under 10 CFR 52.98(f).

Section 52.99(e) NRC inspection, publication of notices, and availability of licensee notifications.

Paragraph (e)(1) of this section indicates that the NRC is responsible for ensuring (through its inspection and audit activities) that the combined license holder performs and documents the completion of inspections, tests, and analyses in the ITAAC. Paragraph (e)(1) requires the NRC to publish, at appropriate intervals until the last date for submission of requests for hearing under 10 CFR 52.103(a), notices in the *Federal Register* of the NRC staff's determination of the successful completion of inspections, tests, and analyses. Paragraph (e)(2) provides that the NRC shall make publicly available the licensee notifications under paragraphs (c) and (f). In general, the NRC expects to make the paragraphs (c) and (f) notifications available shortly after the NRC has received the notifications and concluded that they are complete and detailed. Further, by the date of the *Federal Register* notice of intended operation and opportunity to request a hearing on whether acceptance criteria are met (under

10 CFR 52.103(a)), the NRC will make available the licensee notifications under paragraphs (c)(1), (c)(2), and (c)(3) received to date.

Section 52.99(f) All ITAAC Complete Notification.

Paragraph 52.99(f)(1) would require the licensee to notify the NRC that all ITAAC are complete (All ITAAC Complete Notification). When the licensee submits the all ITAAC complete notification, the NRC would expect that all activities requiring supplemental ITAAC closure letters have been completed, that the associated ITAAC determination bases have been updated, and that all required notifications under paragraphs (c)(3) have been made.

Proposed paragraph (f)(2), which would apply in the period after the licensee has filed the all ITAAC complete notification under paragraph (f)(1), would require the licensee to submit to the NRC a notification that it had identified new information which would require reporting under paragraph (c)(3)(i), that is, that the new information materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. Unlike the 7-day notification requirement under paragraph (c)(3)(i), proposed paragraph (f)(2) would require the licensee to submit the notification no more than 24 hours after the licensee "identifies new information material to the basis" for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. Thus, proposed paragraph (f)(2) would establish a 24-hour limit between the time that the licensee "identifies new information material to" the ITAAC determination basis, and the time that the licensee determines both that the "new information materially alters" the ITAAC determination basis and makes its notification to the NRC.

"New information material to" the ITAAC determination basis is different from, and constitutes a larger set of information than "new information [which] materially alters" the ITAAC determination basis. In the first instance, information "material to" an ITAAC determination

basis constitutes information which a reasonable person with the appropriate education, training, and experience would consider or evaluate in determining whether the ITAAC determination basis is (or continues to be) valid. By contrast, information which "materially alters" the ITAAC determination basis is a subset of the former information. It is information which the same "reasonable person with the appropriate education, training and experience" would conclude – after performing the consideration or evaluation contemplated in the previous criterion – that additional information is necessary to supplement or substitute for, in whole or part, the basis for an ITAAC determination.

The NRC wishes to make clear that once the licensee has filed the all ITAAC complete notification, the 7-day notification requirement under paragraph (c)(3)(i) does not apply. Once the licensee files an all ITAAC complete notification with the NRC, all subsequent licensee determinations that there is new information which "materially alters" the ITAAC determination basis must be filed under paragraph (f)(2).

The NRC is proposing that any 24-hour notifications under paragraph (f)(2) be made to the NRC Operations Center in the same manner as notifications under paragraph (c)(3)(i). The NRC's preferred notification method is by e-mail to hoo.hoc@nrc.gov. Notification can also be made by facsimile to the NRC Operations Center at (301) 816–5151, or by telephone at (301) 816–5100. Verification that the facsimile has been received should be made by calling the NRC Operations Center. Notifications made to the NRC Operations Center under paragraph (f)(2) should clearly indicate that they are being submitted under 10 CFR 52.99(f)(2) and include:

- Name, address, telephone number, and title or position within licensee organization of individual or individuals informing the Commission.

- Identification of the facility reporting the information that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- The date and time that the licensee identified new information material to the basis for determining that a prescribed inspection, test or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- The specific ITAAC affected and the date of the original ITAAC closure notification submitted under 10 CFR 52.99(c)(1).
- The systems affected and the nature of the condition that materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met.
- Whether any corrective actions are planned during the next 24 hours.
- Whether the onsite NRC Resident Inspector has been informed (although the report should include information about whether the Resident Inspector has been informed, this is not a prerequisite for making the report and notification of the Resident Inspector should not inhibit making a timely notification to the NRC Operations Center).

If a 24-hour notification is made under paragraph (f)(2), then the licensee must submit the supplemental ITAAC closure notification under paragraph (c)(3)(ii). All the requirements of paragraph (c)(3)(ii) must be met, including the content and timing of the notification. In addition, the documentation requirements in paragraph (c)(4) apply equally to any paragraph (c)(3)(ii) notifications which may be filed after the all ITAAC complete notification under paragraph (f)(1). Once the licensee has submitted both the 24-hour notification under paragraph (f)(2) and the notification required by paragraph (f)(2) meeting the requirements of paragraph (c)(3)(ii), the licensee must submit a new All ITAAC complete notification in accordance with paragraph (f)(1).

The best way of understanding the regulatory process being proposed in paragraph (f) is

to treat this paragraph as establishing a process for notifications which continues to govern until the Commission makes the 10 CFR 52.103 finding on acceptance criteria. The licensee enters this process when it submits its first all ITAAC complete notification under paragraph (f)(1). Thereafter, the licensee's notification obligations are governed solely by the provisions in paragraph (f), which require the licensee to comply with paragraph (f)(2) and "cycle" back to paragraph (f)(1) until such time as the Commission makes the 10 CFR 52.103(g) finding. The NRC recognizes that paragraph (f)(2) refers back to (c)(3)(ii) with respect to the content and timing of the second notification under paragraph (f)(2), which may lead to the incorrect conclusion that the process is the same before and after the paragraph (f)(1) notification. The NRC may consider additional language changes as part of the final rule to clarify the distinction between these two processes.

V. Guidance

Following issuance of this proposed rule, the NRC will issue a proposed revision to its regulatory guidance in RG 1.215 on implementation of the requirements in 10 CFR 52.99. In this proposed revision, the NRC is considering endorsing Revision 4 to the existing industry ITAAC closure guidance in NEI 08-01, submitted to the NRC for endorsement on July 16, 2010 (ADAMS Accession No. ML102010076). The revised guidance is intended to provide an acceptable method by which licensees can implement the new requirements being proposed in this rulemaking. The staff will consider any comments received on the proposed rule in its final revisions to RG 1.215. The NRC expects that all guidance necessary to implement this rule will be available at the time that the final rule becomes effective.

VI. Specific Request for Comments

In addition to the general invitation to submit comments on the proposed rule, the NRC also requests comments on the following issues:

1. *New information material to the ITAAC determination basis.* Under proposed 10 CFR 52.99(f)(2), which would apply in the period after the licensee has filed the all ITAAC complete notification under paragraph (f)(1), the NRC would require the licensee to submit to the NRC a notification that it had identified new information which would require reporting under proposed 10 CFR 52.99(c)(3)(i), that is, that the new information materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. Unlike the 7-day notification requirement under proposed 10 CFR 52.99(c)(3)(i), proposed 10 CFR 52.99(f)(2) would require the licensee to submit the notification no more than 24 hours after the licensee "identifies new information material to the basis" for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. Thus, proposed 10 CFR 52.99(f)(2) would establish a 24-hour limit between the time that the licensee "identifies new information material to" the ITAAC determination basis, and the time that the licensee determines both that the "new information materially alters" the ITAAC determination basis and makes its notification to the NRC.

"New information material to" the ITAAC determination basis is different from, and constitutes a larger set of information than "new information [which] materially alters" the ITAAC determination basis. In the first instance, information "material to" an ITAAC determination basis constitutes information which a reasonable person with the appropriate education, training, and experience would consider or evaluate in determining whether the ITAAC determination basis is (or continues to be) valid. By contrast, information which "materially alters" the ITAAC determination basis is a subset of the former information. It is information which the same "reasonable person with the appropriate education, training and experience" would conclude – after performing the consideration or evaluation contemplated in the previous criterion – that additional information is necessary to supplement or substitute for, in whole or

part, the basis for an ITAAC determination.

The NRC requests specific comment on whether the NRC should adopt a term other than "material to" for describing the information governing the timing of the 24-hour report, or alternatively, whether the NRC's concept of "material to" needs to be clarified, expanded, or modified.

2. *24-hour reporting time after All ITACC Complete notification.* Under proposed 10 CFR 52.99(f)(1), the licensee would be required to submit an All ITAAC Complete notification. After or concurrent with the last ITAAC closure notification required by 10 CFR 52.99(c)(1), the licensee would be required to notify the NRC that all ITAAC are complete. When the licensee submits the all ITAAC complete notification, the NRC would expect that all activities requiring supplemental ITAAC closure notifications have been completed and that the associated ITAAC determination bases have been updated. However, the NRC recognizes that construction and operational readiness activities will continue even after the licensee submits the all ITAAC complete notification, and that these activities could result in new information that may materially alter the basis for a finding on acceptance criteria under 10 CFR 52.103(g). The NRC understands that prospective licensees expect to complete the last ITAAC very close to the scheduled date for fuel load, and expect the Commission to make the 10 CFR 52.103(g) finding on acceptance criteria shortly thereafter. The NRC's regulatory processes should be structured so that the Commission is able to make a timely 10 CFR 52.103(g) finding that meets all applicable legal standards and is accorded a high level of public confidence. To achieve these objectives, the NRC is proposing provisions in 10 CFR 52.99(f)(2) to address a situation where issues occur after submission of the all ITAAC complete notification. This proposed provision states that if, after filing the all ITAAC complete notification, the licensee identifies new information material to the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met, the licensee

shall determine whether notification is required under 10 CFR 52.99(c)(3)(i) of this section and make the necessary notification within 24 hours of identification of the new information. At this time, the NRC staff would be preparing to make, or may have already made, its recommendation to the Commission in support of the Commission finding under 10 CFR 52.103(g). This prompt notification would be required to ensure that the NRC staff has the most accurate information on which to base its recommendation to the Commission and that the Commission has the most current and accurate information on which to base its finding.

The NRC is seeking specific comment on the time frame for this prompt notification following submittal of the all ITAAC complete notification. The NRC believes that 24 hours is a reasonable amount of time for licensees to evaluate whether new information determined to be material to ITAAC closure will, upon further consideration, materially alter the ITAAC determination basis, given the importance of ensuring that the Commission has complete and accurate information at the time it is determining whether the acceptance criteria in the combined license are met. However, because neither the NRC nor the nuclear power industry have any experience with making these determinations, the NRC is interested in feedback on whether its proposal is reasonable, whether a longer time period should be allowed, or whether no time frame should be specified in the rule itself to allow more detailed guidance on reporting time frames, based on the particular circumstances involved, to be developed in the documents that will provide guidance for implementing these proposed requirements.

VII. Availability of Documents

NRC's Public Document Room (PDR): The public may examine and have copied for a fee publicly available documents at the NRC's PDR, Room O1 F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland.

NRC's Agencywide Documents Access and Management System (ADAMS): Publicly available documents created or received at the NRC are available electronically at the

NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into ADAMS, which provides text and image files of NRC's public documents. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

Federal Rulemaking Web site: Public comments and supporting materials related to this proposed rule can be found at http://www.regulations.gov by searching on Docket ID: NRC-2010-0012.

The NRC is making the documents identified below available to interested persons through one or more of the following methods as indicated.

Document	PDR	Web	ADAMS
SECY-09-0119, "Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses and Acceptance Criteria" (August 26, 2009)	X	X	ML091980372
SRM-M090922 - Staff Requirements - Periodic Briefing on New Reactor Issues - Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC), 9:30 A.M., Tuesday, September 22, 2009 (October 16, 2009)	X	X	ML092890658
Regulatory Guide 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," Revision 0 (October 31, 2009)	X	X	ML091480076
NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3 (January 2009)	X		ML090270415
Regulatory Analysis for Proposed Rule - Requirements for Maintenance of Inspections, Tests, Analyses, and Acceptance Criteria (May 2010)	X	X	ML101440359
NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4 (September 2004)	X	X	ML042820192

VIII. Plain Language

The Presidential memorandum "Plain Language in Government Writing" published June 10, 1998 (63 FR 31883) directed that the Government's documents be in clear and accessible language. The NRC requests comments on the proposed rule specifically with respect to the clarity and effectiveness of the language used. Comments should be sent to the NRC as explained in the ADDRESSES caption of this document.

IX. Agreement State Compatibility

Under the "Policy Statement on Adequacy and Compatibility of Agreement States Programs," approved by the Commission on June 20, 1997, and published in the *Federal Register* (62 FR 46517; September 3, 1997), this rule is classified as compatibility "NRC." Compatibility is not required for Category "NRC" regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the Atomic Energy Act or the provisions of 10 CFR. Although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State's administrative procedure laws. Category "NRC" regulations do not confer regulatory authority on the State.

X. Voluntary Consensus Standard

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104 113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless using such a standard is inconsistent with applicable law or is otherwise impractical. The requirements in this rulemaking address procedural and information collection and reporting requirements necessary to support the NRC's regulatory activities on combined licenses under 10 CFR Part 52, and to facilitate the NRC's conduct of hearings on ITAAC which may be held under Section 189 of the Atomic Energy Act of 1954, as amended. These requirements do not establish standards or substantive requirements with which combined license holders must comply. Thus, this

rulemaking does not constitute establishment of a standard containing generally applicable requirements falling within the purview of the National Technology Transfer and Advancement Act and the implementing guidance issued by the Office of Management and Budget (OMB).

XI. Environmental Impact – Categorical Exclusion

The NRC has determined that these amendments fall within the types of actions described as categorical exclusions under 10 CFR 51.22(c)(2)and (c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this regulation.

XII. Paperwork Reduction Act Statement

This proposed rule contains new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq*). This rule has been submitted to the OMB for review and approval of the information collection requirements.

1. *Type of submission, new or revision:* Revision.
2. *The title of the information collection:* 10 CFR Parts 2 and 52; Requirements for Maintenance of Inspections, Tests, Analyses, and Acceptance Criteria.
3. *Form number, if applicable:* N/A.
4. *How often the collection is required:* On occasion. Reports required under 10 CFR 52.99(c)(3) and (f) are collected and evaluated during construction, (1) whenever a licensee determines that it has new information materially altering the basis for an ITAAC determination; (2) whenever a licensee resolves issues materially altering the basis for an ITAAC determinations; (3) once, when all ITAAC are complete; and (4) whenever a licensee identifies new information which may alter an ITAAC determination basis after it has submitted its notification that all ITAAC are complete.
5. *Who is required or asked to report:* Combined licensee holders, during the period of construction.

6. *An estimate of the number of annual responses:* 99 (88 annual responses plus 3.66 annualized one-time responses plus 7.33 recordkeepers).
7. *The estimated number of annual respondents:* 7.33
8. *The number of hours needed annually to complete the requirement or request:* 13,087 hours (1,210 hours reporting and 11,877 hours recordkeeping).
9. *Abstract:* The NRC is proposing to amend its regulations in 10 CFR 52.99 related to verification of nuclear power plant construction activities through ITAAC under a combined license. Specifically, the NRC is proposing new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure notification. The new provisions would require (1) licensee reporting of new information materially altering the basis for determining that a prescribed inspection, test, or analysis was performed as required, or that a prescribed acceptance criterion is met; (2) licensee documentation of the basis for all ITAAC notifications; and (3) licensee notification of completion of all ITAAC activities.

The NRC is seeking public comment on the potential impact of the information collections contained in this proposed rule and on the following issues:

1. Is the proposed collection of information necessary for the NRC to properly perform its functions? Does the information have practical utility?
2. Is the burden estimate accurate?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the OMB clearance package may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O-1 F21, Rockville, Maryland. The OMB clearance package and rule are available at the NRC worldwide Web site: http://www.nrc.gov/public-involve/doc-comment/omb/index.html for 60 days after the signature date of this notice.

Send comments on any aspect of these proposed regulations related to information collections, including suggestions for reducing the burden and on the above issues, by **(INSERT DATE 30 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*)** to the Information Services Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to Infocollects.Resource@NRC.gov and to the Desk Officer, Christine Kymn, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0151), Office of Management and Budget, Washington, DC 20503. Comments on the proposed information collections may also be submitted via the Federal eRulemaking Portal https://www.regulations.gov, Docket ID NRC-2010-0012. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis

The Commission has prepared a draft regulatory analysis on this proposed regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission.

The Commission requests public comment on the draft regulatory analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading in this document. The analysis is available for inspection in the NRC Public Document Room (ADAMS Accession No. ML101440359), 11555 Rockville Pike, Rockville, Maryland. The analysis may also be viewed and downloaded electronically via the Federal eRulemaking Portal at http://www.regulations.gov by searching for Docket ID NRC-2010-0012.

XIV. Regulatory Flexibility Act Certification

In accordance with the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XV. Backfitting and Issue Finality

The NRC has determined that neither the backfit rule, 10 CFR 50.109, nor any of the finality provisions in 10 CFR Part 52, apply to this proposed rule. Therefore, a backfit analysis is not required because the proposed ITAAC maintenance rule does not contain any provisions that would impose backfitting as defined in the backfit rule, nor does it contain provisions that are inconsistent with the finality provisions applicable to applicants for or holders of combined licenses in 10 CFR Part 52.

The proposed rule would apply only to holders of combined licenses. The backfitting provisions in 10 CFR 50.109 protect holders of combined licenses, and the finality provisions in Subpart C of Part 52 protect holders of combined licenses (with the exception discussed further in this document). There are no current holders of combined licenses; hence, those backfitting

and finality provisions do not apply to this rulemaking. Subpart C of Part 52 contains issue finality provisions which protect combined license applicants, but that protection extends only to issue resolution of matters resolved in referenced early site permits, standard design certifications, standard design approvals, or manufactured reactors. This rule does not alter issue resolution associated with referenced early site permits, standard design certifications, standard design approvals, or manufactured reactors. Instead, this proposed rule addresses requirements concerning the Commission's finding that ITAAC are met, and the conduct of hearings addressing whether prescribed inspections tests and analyses have been performed and the acceptance criteria are met. To the extent that the proposed rule would revise these requirements for future combined licenses, the requirements would not constitute backfitting or otherwise be inconsistent with the finality provisions in 10 CFR Part 52, because the requirements are prospective in nature and effect. Neither the backfit rule nor the issue finality provisions in 10 CFR Part 52 were intended to apply to every NRC action, which substantially changes the obligations of future licensees under 10 CFR Part 52. Accordingly, the NRC has not prepared a backfit analysis or other evaluation for this proposed rule.

List of Subjects

10 CFR Part 2

Administrative practice and procedure, Antitrust, Byproduct material, Classified information, Environmental protection, Nuclear materials, Nuclear power plants and reactors, Penalties, Sex discrimination, Source material, Special nuclear material, Waste treatment and disposal

10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power

plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Parts 2 and 52.

PART 2 - RULES OF PRACTICE FOR DOMESTIC LICENSING PROCEEDINGS AND ISSUANCE OF ORDERS

1. The authority citation for Part 2 continues to read as follows:

AUTHORITY: Secs.161, 181, 68 Stat. 948, 953, as amended (42 U.S.C. 2201, 2231); sec. 191, as amended, Pub. L. 87-615, 76 Stat. 409 (42 U.S.C. 2241); sec. 201, 88 Stat.1242, as amended (42 U.S.C. 5841); 5 U.S.C. 552; sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note).

Section 2.101 also issued under secs. 53, 62, 63, 81, 103, 104, 68 Stat. 930, 932, 933, 935, 936, 937, 938, as amended (42 U.S.C. 2073, 2092, 2093, 2111, 2133, 2134, 2135); sec. 114(f), Pub. L. 97-425, 96 Stat. 2213, as amended (42 U.S.C. 10143(f)); sec. 102, Pub. L. 91-190, 83 Stat. 853, as amended (42 U.S.C. 4332); sec. 301, 88 Stat. 1248 (42 U.S.C. 5871).

Sections 2.102, 2.103, 2.104, 2.105, 2.321 also issued under secs. 102, 103, 104, 105, 183i, 189, 68 Stat. 936, 937, 938, 954, 955, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2233, 2239). Section 2.105 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Sections 2.200-2.206 also issued under secs. 161 b, i, o, 182, 186, 234, 68 Stat. 948-951, 955, 83 Stat. 444, as amended (42 U.S.C. 2201 (b), (i), (o), 2236, 2282); sec. 206, 88 Stat 1246 (42 U.S.C. 5846). Section 2.205(j) also issued under Pub. L. 101-410, 104 Stat. 90, as amended by section 3100(s), Pub. L. 104-134, 110 Stat. 1321-373 (28 U.S.C. 2461 note). Subpart C also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239). Section 2.301 also issued under 5 U.S.C. 554. Sections 2.343, 2.346, 2.712 also issued under 5 U.S.C. 557. Section 2.340 also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161).

Section 2.390 also issued under sec. 103, 68 Stat. 936, as amended (42 U.S.C. 2133) and 5 U.S.C. 552.Sections 2.600-2.606 also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853, as amended (42 U.S.C. 4332). Sections 2.800 and 2.808 also issued under 5 U.S.C. 553. Section 2.809 also issued under 5 U.S.C. 553, and sec. 29, Pub. L. 85-256, 71 Stat. 579, as amended (42 U.S.C. 2039). Subpart K also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97-425, 96 Stat. 2230 (42 U.S.C. 10154). Subpart L also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239). Subpart M also issued under sec. 184 (42 U.S.C. 2234) and sec. 189, 68 Stat. 955 (42 U.S.C. 2239). Subpart N also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239. Appendix A also issued under sec. 6, Pub. L. 91-550, 84 Stat. 1473 (42 U.S.C. 2135).

2. In 10 CFR Part 2.340, paragraph (j) is revised to read as follows:

10 CFR 2.340 Initial decision in certain contested proceedings; immediate effectiveness of initial decisions; issuance of authorizations, permits and licenses.

* * * * *

(j) *Issuance of finding on acceptance criteria under 10 CFR 52.103.* The Commission, the Director of New Reactors, or the Director of Nuclear Reactor Regulation, as appropriate, shall make the finding under 10 CFR 52.103(g) that acceptance criteria in a combined license are met within 10 days from the date of the presiding officer's initial decision:

(1) If the Commission or the appropriate Director is otherwise able to make the finding under 10 CFR 52.103(g) that the prescribed acceptance criteria are met for those acceptance criteria not within the scope of the initial decision of the presiding officer;

(2) If the presiding officer's initial decision—with respect to contentions that the prescribed acceptance criteria have not been met—finds that those acceptance criteria have been met, and the Commission or the appropriate Director thereafter is able to make the finding that those acceptance criteria are met;

(3) If the presiding officer's initial decision—with respect to contentions that the prescribed acceptance criteria will not be met—finds that those acceptance criteria will be met, and the Commission or the appropriate Director thereafter is able to make the finding that those acceptance criteria are met; and

(4) Notwithstanding the pendency of a petition for reconsideration under 10 CFR 2.345, a petition for review under 10 CFR 2.341, or a motion for stay under 10 CFR 2.342, or the filing of a petition under 10 CFR 2.206.

* * * * *

PART 52 - LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

3. The authority citation for Part 52 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 185, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2235, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109-58, 119 Stat. 594 (2005), Secs. 147 and 149 of the Atomic Energy Act.

4. Section 52.99 is revised to read as follows:

10 CFR 52.99 Inspection during construction; ITAAC schedules and notifications; NRC notices.

(a) *Licensee schedule for completing inspections, tests or analyses.* The licensee shall submit to the NRC, no later than 1 year after issuance of the combined license or at the start of construction as defined at 10 CFR 50.10(a), whichever is later, its schedule for completing the inspections, tests, or analyses in the ITAAC. The licensee shall submit updates to the ITAAC schedules every 6 months thereafter and, within 1 year of its scheduled date for initial loading of

fuel, the licensee shall submit updates to the ITAAC schedule every 30 days until the final notification is provided to the NRC under paragraph (c)(1) of this section.

(b) *Licensee and applicant conduct of activities subject to ITAAC.* With respect to activities subject to an ITAAC, an applicant for a combined license may proceed at its own risk with design and procurement activities, and a licensee may proceed at its own risk with design, procurement, construction, and preoperational activities, even though the NRC may not have found that any one of the prescribed acceptance criteria are met.

(c) *Licensee notifications and documentation.*

(1) *ITAAC closure notification.* The licensee shall notify the NRC that prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met. The notification must contain sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the prescribed acceptance criteria are met.

(2) *Uncompleted ITAAC notification.* If the licensee has not provided, by the date 225 days before the scheduled date for initial loading of fuel, the notification required by paragraph (c)(1) of this section for all ITAAC, then the licensee shall notify the NRC that the prescribed inspections, tests, or analyses for all uncompleted ITAAC will be performed and that the prescribed acceptance criteria will be met prior to operation. The notification must be provided no later than the date 225 days before the scheduled date for initial loading of fuel, and must provide sufficient information to demonstrate that the prescribed inspections, tests, or analyses will be performed and the prescribed acceptance criteria for the uncompleted ITAAC will be met, including, but not limited to, a description of the specific procedures and analytical methods to be used for performing the prescribed inspections, tests, and analyses and determining that the prescribed acceptance criteria are met.

(3) *ITAAC post-closure notifications.* The requirements in this paragraph apply, with

respect to each ITAAC, after the licensee makes an ITAAC closure notification under paragraph (c)(1) of this section

(i) *New information on ITAAC closure.* The licensee shall notify the NRC of new information materially altering the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. The notification shall be by e-mail to hoo.hoc@nrc.gov, which is the preferred method of notification, by facsimile to the NRC Operations Center at (301) 816–5151, or by telephone at (301) 816–5100 within 7 days following licensee determination that the new information materially alters the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met. Verification that the e-mail or facsimile has been received should be made by calling the NRC Operations Center.

(ii) *Supplemental ITAAC closure notification.* The licensee shall notify the NRC of its resolution of issues reported under paragraph (c)(3)(i) of this section. The notification must contain sufficient information to demonstrate that, notwithstanding the new information, the prescribed inspections, tests, or analyses have been performed as required, and the prescribed acceptance criteria are met. The notification must be made no later than 30 days after licensee resolution of the issue.

(4) *ITAAC closure documentation.* The licensee shall maintain records of the bases for determining whether a notification under paragraph (c)(3)(i) of this section is required and the bases for all notifications made under paragraph (c) of this section. The licensee shall retain these records for 5 years after the date the Commission makes the finding under 10 CFR 52.103(g).

(d) *Licensee determination of non-compliance with ITAAC.*

(1) In the event that an activity is subject to an ITAAC derived from a referenced

standard design certification and the licensee has not demonstrated that the prescribed acceptance criteria are met, the licensee may take corrective actions to successfully complete that ITAAC or request an exemption from the standard design certification ITAAC, as applicable. A request for an exemption must also be accompanied by a request for a license amendment under 10 CFR 52.98(f).

(2) In the event that an activity is subject to an ITAAC not derived from a referenced standard design certification and the licensee has not demonstrated that the prescribed acceptance criteria are met, the licensee may take corrective actions to successfully complete that ITAAC or request a license amendment under 10 CFR 52.98(f).

(e) *NRC inspection, publication of notices, and availability of licensee notifications.* The NRC shall ensure that the prescribed inspections, tests, and analyses in the ITAAC are performed.

(1) At appropriate intervals until the last date for submission of requests for hearing under 10 CFR 52.103(a), the NRC shall publish notices in the *Federal Register* of the NRC staff's determination of the successful completion of inspections, tests, and analyses. If such a notice is published and the licensee notifies the NRC in accordance with paragraph (c)(3)(i) of this section before the last date of submission of requests for hearing, then the NRC will, until the last date for submission of requests for hearing under 10 CFR 52.103(a), publish notices in the *Federal Register* of the licensee's submission of a notification under paragraph (c)(3)(i) of this section and any NRC staff determination that the acceptance criteria for the affected ITAAC are met.

(2) The NRC shall make publicly available the licensee notifications under paragraphs (c) and (f) of this section. The NRC shall make publicly available the licensee notifications under paragraphs (c)(1), (c)(2), and (c)(3) of this section no later than the date of publication of the notice of intended operation required by 10 CFR 52.103(a).

(f) *All ITAAC Complete notification.*

(1) The licensee shall notify the NRC that all ITAAC are complete.

(2) If, after making the notification required under paragraph (f)(1) of this section, the licensee identifies new information material to the basis for determining that a prescribed inspection, test, or analysis was performed as required or finding that a prescribed acceptance criterion is met, the licensee shall determine whether notification is required under paragraph (c)(3)(i) of this section and make the necessary notification within 24 hours of identification of the new information. The notification shall be to the NRC Operations Center in the same manner as notifications made under paragraph (c)(3)(i) of this section. The licensee must submit any notifications required under paragraph (c)(3)(ii) of this section and resubmit the notification required by paragraph (f)(1) of this section.

Dated at Rockville, Maryland, this day of 2010.

For the Nuclear Regulatory Commission

Annette L. Vietti-Cook,
Secretary of the Commission.

Regulatory Analysis for Proposed Rule - Requirements for Maintenance of Inspections, Tests, Analyses, and Acceptance Criteria

U.S. Nuclear Regulatory Commission

May 2010

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations related to verification of nuclear power plant construction activities through inspections, tests, analyses, and acceptance criteria (ITAAC) under a combined license. Specifically, the NRC is proposing new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure notification. The new provisions would require licensees to report new information materially altering the basis for determining that a prescribed inspection, test, or analysis was performed as required, or finding that a prescribed acceptance criterion is met; to document the basis for all ITAAC notifications; and to notify the NRC of completion of all ITAAC activities. In addition, the NRC is proposing editorial corrections to existing language in the NRC's regulations to correct and clarify ambiguous language and make it consistent with language in the Atomic Energy Act of 1954, as amended (AEA).

The analysis presented in this document examines the benefits and costs of the proposed requirements. The key findings of the analysis are as follows:

- *Total Cost to Industry*. The proposed rule would result in additional reporting and recordkeeping costs for the industry. The total annual cost for the rule is $2,013,480. The total present value of the costs is estimated at $37,668,316 (using a 7-percent discount rate) and $39,177,089 (using a 3-percent discount rate) over the next 20 years.

- *Annual Impact to the Economy*. Under the Congressional Review Act of 1996 and as a result of consultations with the Office of Information and Regulatory Affairs of the Office of Management and Budget, the NRC has determined that this action is a non-major rule. This determination is based on the estimated one-time costs (expected to occur within the first year) of implementing this action for the total industry is not to exceed $274,555.

- *Value of Benefits Not Reflected Above*. The cost figures shown above do not reflect the value of the benefits of the proposed rule. These benefits are evaluated qualitatively in Section 3.1. This regulatory analysis concluded the costs of the rule are justified in view of the qualitative benefits.

- *Costs to NRC*. The annual cost of the rule to the NRC is negligible. The NRC would incur costs to review and process licensee responses to the proposed reporting requirements and to conduct inspections triggered by the new notifications. The total annual costs are approximately $509,184. The NRC will incur one-time costs for developing the infrastructure to process the new notifications, developing guidance, and training NRC staff on the proposed requirements estimated to be $63,360.

- *Decision Rationale*. Although the NRC did not quantify the benefits of this rule, the staff did qualitatively examine benefits and concluded that the rule would provide enhanced regulatory effectiveness and efficiency and enhanced openness of the regulatory process. The sum total of the requirements in the proposed rule would be to establish reporting of issues affecting closed ITAAC. Specifically, the proposed rule would require the following:

(1) licensee reporting of new information materially altering the basis for determining that a prescribed inspection, test or analysis was performed as required, or finding that a prescribed acceptance criterion is met;

(2) licensee documentation of the basis for all ITAAC notifications; and

(3) licensee notification of completion of all ITAAC activities.

The proposed amendments would affect NRC licensees who have received a combined license and who have begun construction.

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ACRONYMS AND ABBREVIATIONS

AEA	Atomic Energy Act of 1954, as amended
ADAMS	Agencywide Documents Access and Management System
CFR	*Code of Federal Regulations*
FR	*Federal Register*
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
NRC	Nuclear Regulatory Commission
NUREG/BR	NRC Nuclear Regulation/Brochure
RG	Regulatory Guide

1. INTRODUCTION

The NRC is proposing to amend the Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 2 and 52 to specify additional requirements for reporting and record keeping of ITAAC activities. The NRC regulations in 10 CFR 52.99 require NRC licensees to provide an ITAAC closure letter once a licensee has successfully completed a required inspection, test, or analysis and determined that the associated acceptance criteria are met. This rulemaking would amend these regulations to identify other occasions where a notification of activities affecting closed ITAAC would be required. The proposed rule would also require licensee documentation of the bases for all ITAAC notifications required under 10 CFR 52.99(c), as well as make corrections to existing language in 10 CFR 52.99 for consistency with other sections in 10 CFR Part 52 and with language in the Atomic Energy Act, as amended (the Act). Finally, this rulemaking would amend 10 CFR 2.340(j) to correct errors and clarify ambiguous statements.

This regulatory analysis has been prepared in accordance with the Regulatory Analysis Guidelines (RA Guidelines) of the NRC, NUREG/BR-0058, Revision 4, September 2004. This regulatory analysis evaluates the consequences associated with the "Requirements for Maintenance of Inspections, Tests, Analyses, and Acceptance Criteria" proposed rule. This document presents background material, rulemaking objectives, alternatives, input assumptions, and analysis of the consequences of the rule language. The regulatory analysis consists of two parts. The first is an aggregate analysis of the proposed rule. The second part is a screening review for disaggregation to identify any individual provisions whose costs are disproportionate to the potential benefits.

The remainder of this introduction is divided into two sections. Section 1.1 states the problem and the objective of the rulemaking. Section 1.2 provides background information and Section 2 identifies the alternatives evaluated in this rulemaking. Section 3 describes the analysis method and input assumptions, Section 4 describes the results, Section 5 discusses the decision rationale, Section 6 the Implementation of the preferred alternative, and Section 7 lists the references used in this Regulatory Analysis.

1.1 Statement of the Problem

As the NRC developed its processes for verification of nuclear power plant construction activities through ITAAC under a combined license, it became clear that there were a number of implementation issues left unaddressed by the existing provisions in 10 CFR Part 52. In particular, the NRC believes that additional notifications should be provided to the NRC by the combined license holder following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1). In general, the reasons for these proposed new notifications are to ensure that the NRC has sufficient information, in light of new information developed or identified after ITAAC completion and NRC notification, to complete all of the activities necessary for the Commission to make a determination on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing. Therefore, the NRC is proposing new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure letter.

1.2. Background

1.2.1 Current Regulatory Framework

In 54 FR 15371 (April 18, 1989), the Commission added 10 CFR 52.99, "Inspection during construction," to clearly reflect that inspections (carried out during construction under a combined license) would be based on ITAAC proposed by the applicant, approved by the staff, and incorporated in the combined license. At that time, the Commission made it clear that it would make no findings with respect to construction until the construction was complete. Nonetheless, 10 CFR 52.99 envisioned a "sign-as-you-go" process, whereby NRC staff signed-off on inspection units and notice of the staff's sign-off would be published in the *Federal Register.*

In 2007, the Commission revised 10 CFR Part 52 to enhance the NRC's license implementation and approval processes (72 FR 49351; August 28, 2007). In that revision, the NRC amended 10 CFR 52.99 to require licensees to notify the NRC that the prescribed inspections, tests, and analyses in the ITAAC were complete and that the acceptance criteria were met. The revision also required that notifications sufficiently demonstrate that the prescribed inspections, tests, and analyses were performed and the prescribed acceptance criteria were met. The NRC added this requirement to ensure that combined license applicants and holders were aware that (1) it was the licensee's burden to demonstrate compliance with the ITAAC and (2) the NRC expected the notification of ITAAC completion to contain more information than just a simple statement that the licensee believes the ITAAC had been completed and the acceptance criteria met.

Under Section 185.b of the AEA and 10 CFR 52.97(b), a combined license for a nuclear power plant must contain ITAAC that are "necessary and sufficient to reasonably assure that the facility was constructed and will operate in conformity with" the license, the AEA, and NRC regulations. Following issuance of the combined license, Section 189.b of the AEA and 10 CFR 52.99(e) require that the Commission "ensure that the prescribed inspections, tests, and analyses are performed." Finally, before operation of the facility, Section 189.b and 10 CFR 52.103(g) require that the Commission find that the "prescribed acceptance criteria are met." This Commission finding will not occur until construction is complete, near the date for scheduled initial fuel load.

As currently required by 10 CFR 52.99(c)(1), the licensee must submit ITAAC closure letters containing "sufficient information to demonstrate that the prescribed inspections, tests, and analyses have been performed and that the associated acceptance criteria have been met." These notification letters perform two functions, as discussed in the Supplementary Information for the 2007 final rule amending Part 52: (1) T hey alert the NRC to the licensee's ITAAC completion and ensure that the NRC has sufficient information to complete all of the necessary activities for the Commission to make a determination as to whether all of the ITAAC have been or will be met (the latter is relevant to any hearing on ITAAC under 10 CFR 52.103) before initial operation; and (2) They ensure that interested persons have access to completed and uncompleted ITAAC information at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing on acceptance criteria (72 FR 49352; August 28, 2007, at 49450 (second column)).

Following the 2007 rulemaking, the NRC began to develop ITAAC closure process guidance on the requirements under 10 CFR 52.99. In October 2009, the NRC issued regulatory guidance for the implementation of the revised 10 CFR 52.99 in Regulatory Guide (RG) 1.215, "Guidance

for ITAAC Closure Under 10 CFR Part 52," which endorsed guidance developed by the Nuclear Energy Institute (NEI) in NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3, issued January 2009 (ADAMS Accession No. ML090270415).

The NRC realized (after a series of public meetings) that a number of additional implementation issues were left unaddressed by various provisions found in 10 CFR Part 52. In particular, the NRC believes that additional notifications should be provided to the NRC by the combined license holder following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1).

1.2.2 Regulatory Objectives

The NRC's objectives for the proposed rulemaking are to: (1) establish a new provision requiring licensees to report new information that materially alters the basis for determining that a prescribed inspection, test or analysis was performed as required, or finding that a prescribed acceptance criterion was met; (2) establish a new provision requiring licensees to document the basis for all ITAAC notifications; (3) establish a new provision to require licensees to notify NRC when all ITAAC activities are complete; and (4) make corrections to existing language in 10 CFR 2.340 and 52.99 to be consistent with other sections in 10 CFR Part 52 and with language in the AEA.

2. IDENTIFICATION OF ALTERNATIVE APPROACHES

The following discussion describes the two regulatory options being considered, with additional analysis presented in Section 3.

2.1 Alternative 1: No-Action

Under Option 1, the No-action alternative, NRC would not amend the current regulations regarding additional ITAAC notifications. The NRC would continue to work with industry to develop regulatory guidance to achieve the NRC's goals. This option would avoid certain costs that the rule would impose. However, taking no action would not ensure that the NRC has sufficient information, in light of new information developed or identified after ITAAC completion, to complete all of the activities necessary for the Commission to make a determination on ITAAC, as required by the AEA, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the threshold for requesting a hearing. The baseline of the analysis is Option 1, the No-action alternative, for which there are no costs.

2.2 Alternative 2: Rulemaking to Amend Regulations to Add ITAAC Notification and Recordkeeping Requirements

Under this option, NRC would conduct a rulemaking to amend its regulations in 10 CFR Part 52 related to verification of nuclear power plant construction activities through inspections, tests, analyses, and acceptance criteria (ITAAC) under a combined license. These changes are to: (1) amend 10 CFR 52.99(c)(3) to require licensee reporting of new information materially altering the basis for determining that a prescribed inspection, test or analysis was performed as required, or finding that a prescribed acceptance criterion is met; (2) amend 10 CFR 52.99(f) to require licensee notification of completion of all ITAAC activities, (3) amend 10 CFR 52.99(c)(4) to require licensee documentation of the basis for all ITAAC notifications, and (4) make

corrections to existing language in 10 CFR 2.340 and 52.99 to be consistent with other sections in 10 CFR Part 52 and with language in the AEA.

This alternative would be consistent with NRC's organizational excellence objectives of ensuring that its actions are efficient, effective, realistic, and timely. The rulemaking alternative is more efficient and effective than relying on voluntary actions by licensees to notify the NRC of these events. It would also be consistent with NRC's openness strategy. This alternative, through the rulemaking process, would provide for fair, timely, and meaningful stakeholder involvement in NRC's development of its ITAAC closure process.

The NRC has estimated the benefits and costs of this alternative, as described in Sections 3 and 4 of this regulatory analysis. The NRC has pursued Alternative 2: Rulemaking for the reasons discussed in Section 5.

3. ESTIMATION AND EVALUATION OF VALUES AND IMPACTS

3.1 Identification of Affected Attributes

This section describes the analysis of private and public sector factors that the proposed rule is expected to affect. The analysis is conducted to identify and evaluate the benefits (values) and costs (impacts) of the two regulatory options, using the list of potential attributes provided in Chapter 5 of NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," dated January 1997, and in Chapter 4 of NUREG/BR-0058, Rev. 5, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," dated September 2004. The evaluation considered each attribute listed in Chapter 5. The basis for selecting those attributes is presented below. Section 3.1 identifies the attributes expected to be affected by the proposed rulemaking. Section 3.2 describes how the values and impacts have been analyzed. Finally, Section 3.3 presents the detailed results of the projected values and impacts.

Affected attributes include the following:

- *Industry Implementation* --The regulatory action would result in the need for combined license holders to read the amended regulations and develop procedures for processing notifications required by the proposed new provisions. The regulatory action would also require licensees to develop procedures for documenting the basis for all ITAAC notifications required under 10 CFR 52.99(c).

- *Industry Operation* -- The regulatory action would require combined license holders to: (1) report new information materially altering the basis for determining that a prescribed inspection, test or analysis was performed as required or finding that a prescribed acceptance criterion is met and (2) notify the NRC of completion of all ITAAC activities. The regulatory action would also require combined license holders to retain records of the bases for all ITAAC notifications required under 10 CFR 52.99(c) throughout construction and for five years after the date the Commission makes the finding under 10 CFR 52.103(g) that will allow fuel load and operation.

- *NRC Implementation* – The NRC would incur costs to develop rule guidance, develop the infrastructure to process the proposed new

notifications, develop inspection procedures for inspection activities triggered by the proposed new notifications, and develop and conduct NRC staff training on the new requirements.

- *NRC Operation* -- Under the regulatory actions, the NRC would incur costs to review licensee responses to the new reporting requirements of the proposed rule and to perform additional inspections as a result of the new reports.

- *Improvements in Knowledge* – The regulatory action would improve knowledge with regard to activities affecting closed ITAAC at facilities under construction.

- *Regulatory Efficiency* -- The regulatory action would improve regulatory efficiency by ensuring that the NRC has sufficient, timely information, in light of new information developed or identified after ITAAC completion and NRC notification, to complete all of the activities necessary for the Commission to make a determination on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing.

- *General Public* -- The regulatory action would improve the general public's ability to participate effectively in the licensing process by ensuring that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189a(1)(B) threshold for requesting a hearing.

Attributes that are *not* expected to be affected by the rulemaking options include the following:

- Occupational Health (Routine);
- Occupational Health (Accident);
- Public Health (Routine);
- Public Health (Accident);
- Off-site Property;
- On-site Property;
- Environmental Considerations;
- Antitrust Considerations;
- Other Government;
- Safeguards and Security Considerations

3.2 Analytical Methodology

This section describes the methodology used to analyze the consequences associated with the proposed rule. The values (benefits) include any desirable changes in the affected attributes. The impacts (costs) include any undesirable changes in affected attributes.

The NRC collected input assumptions using data and information from the following sources: NRC workgroups and staff experience; NRC databases; and reports and documents.

As described in Section 3.1, the attributes expected to be affected include the following:

- Industry Implementation
- Industry Operation

- NRC Implementation
- NRC Operation
- Improvements in Knowledge
- Regulatory Efficiency
- General Public

This analysis relies on a qualitative evaluation for several of the affected attributes (e.g., improvements in knowledge, regulatory efficiency, and general public) due to difficulty in quantifying the impact of the current rulemaking. The remaining attributes (industry implementation, industry operation, NRC implementation, and NRC operation) are evaluated quantitatively. The analysis proceeds quantitatively for these attributes and makes assumptions as discussed in Section 3.2.1.

In accordance with Office of Management and Budget guidance and NUREG/BR-0058, Rev. 4, the results of the analysis are presented using both 3 percent and 7 percent real discount rates. The NRC seeks public comments on the accuracy of these regulatory analysis assumptions and on the validity of the proposed rules value and impact estimation methods.

3.2.1 Data and Assumptions

3.2.1.1 Affected Entities

Licensees

This regulatory action would affect combined license holders who have begun ITAAC closure activities. The NRC estimates that this could affect 17 licensees over the next 20 years, based on the published schedules for combined license applications currently under NRC review.

NRC

NRC costs for implementing this regulation would be incurred primarily by the Office of New Reactors. There would also be costs incurred by Region II for additional inspections.

3.2.1.2 Other Data and Attributes

- Assumed labor rate for NRC staff is $120 per hour and for licensee personnel is $100 per hour.

- Ongoing costs of operation related to the rule are assumed to begin in 2011, and are modeled on an annual cost basis. Ongoing costs related to the No-Action Alternative are assumed to be ongoing and begin in 2011 and are modeled on an annual cost basis.

- The analysis calculated cost and savings over a 4-year construction timeframe and a 3-year post-construction recordkeeping period, with each year's costs or savings discounted back at a 7-percent and 3-percent discount rate, in accordance with NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Rev. 4.

- For the analysis, annual costs have been multiplied by 1.2 to capture inflationary cost increases incurred by the 17 licensees constructing over a 20 year staggered time-frame.

- For the analysis, the NRC assumed that all combined license applications currently under active review would be approved and issued on their current published schedules. In addition, the NRC assumed that each combined license holder would begin construction upon issuance of the combined license and that construction would span a period of 4 years. The NRC also assumed that each licensee would submit 6 supplemental ITAAC closure notifications per year of construction and that 2 of those supplemental notifications would result in additional NRC inspection.

- Finally, for analysis of the recordkeeping requirements, the NRC assumed that, on average, ITAAC closure records would need to be retained for 7 years (2 years during construction and 5 years after the 10 CFR 52.103(g) finding).

3.3 Detailed Results

This section presents a detailed estimate of the values and impacts for the proposed rulemaking (Option 2). Some values and impacts are addressed qualitatively for reasons discussed in Section 3.2. These results are summarized in Table 3.

Option 1: No-action

By definition, this option does not result in any values or impacts.

Option 2: Amend Regulations to Add ITAAC Notification and Recordkeeping Requirements

Industry Implementation

Impact: Read the amended regulations.

- One time incremental effort of 1.5 hours per licensee.

Impact: Develop procedures for processing notifications required by the proposed new provisions.

- One time incremental effort of 80 hours per licensee.

Impact: Develop procedures for documenting the basis for all ITAAC notifications required under 10 CFR52.99(c).

- One time incremental effort of 80 hours per licensee.

Industry Operation

Impact: Report new information materially altering the ITAAC determination basis.

- Effort of 1.5 hours per licensee for each early notification under

- 10 CFR 52.99(c)(3)(i).
- Effort of 20 hours per licensee for each supplemental ITAAC closure letter under 10 CFR 52.99(c)(3)(ii).

Impact: Submit All ITAAC Complete Letter under 10 CFR 52.99(f).

- Effort of 8 hours per licensee.

Impact: Recordkeeping:

- 20 hours per ITAAC for each licensee to develop records documenting the basis for all ITAAC notifications required under 10 CFR 52.99(c) and 1 hour per year per ITAAC for each licensee to maintain records (810 hours per year per licensee).

NRC Implementation

Impact: Develop rule guidance:

- One time incremental effort of 200 hours to develop new guidance or revise existing guidance.

Impact: Develop infrastructure to process supplemental ITAAC notifications and All ITAAC Complete notifications:

- One time incremental effort of 160 hours of labor.

Impact: Develop inspection procedures for inspection of activities triggered by early notification of ITAAC maintenance issues under 10 CFR 52.99(c)(3)(i):

- One time incremental effort of 80 hours.

Impact: Develop and conduct NRC staff training on new requirements.

- One time incremental effort of 40 hours.

NRC Operation

Impact: Review and process early notifications required by 10 CFR 52.99(c)(3)(i):

- Incremental effort of 8 hours per report to collect, review, and process early notifications.

Impact: Review and process supplemental ITAAC closure notifications under 10 CFR 52.99(c)(3)(ii):

- Incremental effort of 20 hours per report.

Impact: Perform unplanned inspection of ITAAC maintenance issues:

- Annual incremental effort of 20 hours per licensee.

Impact: Review and process All ITAAC Complete notification under 10 CFR 52.99(f)(1)

- Incremental effort of 8 hours per report to collect, review, and process.

Table 3
Quantitative Results
Value (+) or Impact (-)

	One-time Implementation Costs	*Annual Operating Costs
Industry Costs	**$274,555	$2,013,480
NRC Costs	$63,360	$509,184
Total	$337,915	$2,522,664

*Annual Operating costs have been factored by 1.2 to account for inflation over the 20 year construction period that the 17 licensees will be constructing plants.
**The one-time industry reporting cost of $27,540,000 is not reflected here, because it is not an implementation cost.

4. PRESENTATION OF RESULTS

4.1 Values and Impacts

This section presents results of values and impacts (i.e., costs) that are expected to be derived from the proposed rule. To the extent that the affected attributes could be analyzed quantitatively, the net effect of each alternative has been calculated and is presented below. However, some values and impacts could be evaluated only on a qualitative basis.

The results of the value-impact analysis are summarized in Tables 4-1 and 4-2. Table 4-3 provides the cost comparison for the two alternatives. The Rulemaking Alternative would result in additional costs when compared to the No-Action Alternative. The quantitative impact estimated for the Rulemaking Alternative is in the millions. The rulemaking is estimated to cost between $39,443,488 and $41,092,239 (7-percent and 3-percent discount rate, respectively). Costs are mostly borne by industry.

TABLE 4-1

Summary of Benefits/Savings and Costs/Burdens

Net Monetary Savings (or Costs) – Total Present Value in millions	Non-Monetary Benefits/Costs
Alternative 1: No Action **Industry:** $0 **NRC:** $0	Qualitative Benefits: None. Qualitative Costs: Regulatory Efficiency: Regulatory efficiency would be reduced by not providing the most efficient timely ITAAC completion notifications. General Public: The general public's ability to participate effectively in the licensing process could be reduced because taking no action would not ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing.
Alternative 2: Rulemaking **Industry:** ($39.46) using a 3% discount rate ($37.94) using a 7% discount rate **NRC:** ($1.63) using a 3% discount rate ($1.50) using a 7% discount rate	Qualitative Benefits: Improvements in Knowledge: Increase knowledge of closed ITAAC at facilities under construction. Regulatory Efficiency: Improve regulatory efficiency by ensuring that the NRC has sufficient, timely information, in light of new information developed or identified after ITAAC completion and NRC notification. General Public: Improve the general public's ability to participate effectively in the licensing process. Qualitative Costs: None.

Table 4-2 presents the net impact of the rule. A positive value below is a cost. A number in parentheses is a negative cost, or a savings.

Table 4-2: Net Impact of Alternatives 1 and 2

Regulatory Alternative	Total at 3% discount rate ($)	Total 7% discount rate ($)
1. No-Action	$0	$0
2. Rulemaking	$41,092,239	$39,443,488

*Reporting was recorded over a 4 year timeframe and recordkeeping over a 7 year timeframe. This was factored by 1.20 for inflationary considerations (see attributes section for further information).

There are no "new" substantial costs to industry associated with the No-Action Alternative. No changes would be made to the regulation.

There are no quantifiable values (i.e. benefits) associated with the rule. The qualitative values of the rule are associated with improved regulatory efficiency by ensuring that the NRC has sufficient, timely information, in light of new information developed or identified after ITAAC completion and NRC notification, to complete all of the activities necessary for the Commission to make a determination on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing. This has a beneficial effect on the attributes of improvements in knowledge, regulatory efficiency, and general public.

Table 4-3 shows the estimated costs by attribute.

Table 4-3: Estimated Values and Impacts by Attribute

Attribute	Alternative 2: Rulemaking Total Cost (million $)	
	3% Discount	7% Discount
Industry Implementation	(.27)	(.27)
Industry Operation	(39.19)	(37.67)
NRC Implementation	(.06)	(.06)
NRC Operation	(1.57)	(1.44)
Total	(41.09)	(39.44)

Note: Total may differ from sum of values due to rounding.
*4 years have been considered for ITAAC reporting and 7 years for ITAAC recordkeeping purposes

5. DECISION RATIONALE

NRC's current regulations in 10 CFR 52.99 require licensees to notify the NRC that the prescribed inspections, tests, and analyses in the ITAAC were complete and that the acceptance criteria were met. As the NRC developed its processes for verification of nuclear power plant construction activities through ITAAC under a combined license, it became clear that there were a number of implementation issues left unaddressed by the existing provisions in 10 CFR Part 52. In particular, the NRC believes that additional notifications should be provided to the NRC by the combined license holder following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1). In general, the reasons for these proposed new notifications are to ensure that the NRC has sufficient information, in light of new information developed or identified after ITAAC completion and NRC notification, to complete all of the activities necessary for the Commission to make a determination on ITAAC, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing. Therefore, the NRC is proposing new provisions that apply after a licensee has completed an ITAAC and submitted an ITAAC closure letter.

5.1 Aggregate Analysis

Two alternatives were evaluated in this Regulatory Analysis. Alternative 1, the No-Action Alternative, would maintain the regulations as currently written and the NRC would continue to work with industry to develop regulatory guidance to achieve the NRC's goals. This option would avoid certain costs that the rule would impose. However, taking no action would not ensure that the NRC has sufficient information, in light of new information developed or identified after ITAAC completion, to complete all of the activities necessary for the Commission to make a determination on ITAAC, as required by the AEA, and to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the threshold for requesting a hearing.

Alternative 2, the Rulemaking Alternative, would amend NRC regulations to: (1) amend 10 CFR 52.99(c)(3) to require licensee reporting of new information, materially altering the basis for determining that a prescribed inspection, test or analysis was performed as required, or finding that a prescribed acceptance criterion is met; (2) amend 10 CFR 52.99(f) to require licensee notification of completion of all ITAAC activities, (3) amend 10 CFR 52.99(c)(4) to require licensee documentation of the basis for all ITAAC notifications, and (4) make corrections to existing language in 10 CFR 2.340 and 52.99 to be consistent with other sections in 10 CFR Part 52 and with language in the Act. Alternative 2 would improve regulatory efficiency by ensuring that the NRC has sufficient, timely information, in light of new information developed or identified after ITAAC completion and NRC notification, to complete all of the activities necessary for the Commission to make a determination on ITAAC, and ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189a(1)(B) threshold for requesting a hearing. Therefore, the Rulemaking Alternative is the preferred approach.

5.2 Disaggregation Analysis

The NRC has prepared an analysis of the impact of the changes (Appendix A) that identifies each provision affected by the rulemaking and determines its contribution to the overall cost of the proposed rule. The NRC has determined that each individual requirement is needed for the regulatory initiative to resolve the problems and concerns and meet the stated objectives that are the focus of the regulatory initiative, as illustrated in Table 5-1 below. The NRC also performed an analysis to identify any individual provision that could impose cost disproportionate to the benefits attributable to each provision. The NRC has concluded that there are no provisions whose costs are disproportionate to the benefits and whose inclusion in the aggregate analysis could mask the impact of this rulemaking.

Table 5-1: Disaggregation Analysis

Rule Objectives	52.99(c)(3)(i)	52.99(c)(3)(ii)	52.99(f)(1)	52.99(f)(2)	52.99(c)(4)
	New information on ITAAC closure	Supplemental ITAAC closure notification	All ITAAC Complete notification	24-hour notification of new information	ITAAC closure records
NRC has information to support inspections	X			X	X
NRC has sufficient information for ITAAC finding	X	X	X	X	X
Interested persons have access to ITAAC information		X			

6. IMPLEMENTATION

The staff is recommending that the final rule be effective 30 days after publication in the *Federal Register*. The industry has proactively been revising their own guidance document to require many of the things that would be imposed by this proposed rule. Therefore, the NRC expects that combined license holders will already be planning for these reports and records by the time this rule is promulgated, should it be adopted by the NRC in a final rule.

7. REFERENCES

NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Rev. 4.

NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook, Final Report," Office of Nuclear Regulatory Research, January 1997.

SECY-09-0119, "Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses and Acceptance Criteria," August 26, 2009.

SRM-M090922 - "Staff Requirements - Periodic Briefing on New Reactor Issues - Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC), 9:30 A.M., Tuesday, September 22, 2009," October 16, 2009.

Regulatory Guide 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," Revision 0, October 31, 2009.

NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3, January 2009."

Appendix: Analysis Details

52.99(c)(3)(i) New Information on ITAAC Closure.

NRC's current regulations in 10 CFR 52.99 require licensees to notify the NRC that the prescribed inspections, tests, and analyses in the ITAAC were complete and that the acceptance criteria were met. The NRC believes that additional notifications should be provided to the NRC by the combined license holder following the notification of ITAAC completion currently required by 10 CFR 52.99(c)(1). The proposed revisions would require licensees to notify the NRC of new information materially altering the basis for determining that a prescribed inspection, test or analysis was performed as required or finding that a prescribed acceptance criterion is met. The notification would be required to be by e-mail to hoo.hoc@nrc.gov, which is the preferred method of notification, by facsimile to the NRC Operations Center at (301) 816–5151, or by telephone at (301) 816–5100 within 7 days following licensee determination that the new information materially alters the basis for determining that a prescribed inspection, test or analysis was performed as required or finding that a prescribed acceptance criterion is met.

NRC Costs to Review Early Notifications

Cost of NRC staff time	$120
Hours of NRC staff time	x 8
Average Annual No. of ITAAC per Licensee	x 6
Number of Licensees	x 17
Total Annual Early Notification Review Costs	**($97,920)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
TOTAL NRC EARLY NOTIFICATION REVIEW COSTS	**($470,016)**

NRC Costs to Perform Unplanned ITAAC Inspections

Cost of NRC staff time	$120
Hours of NRC staff time	x 20
Average Annual No. of ITAAC per Licensee	x 2
Number of Licensees	x 17
Total Annual Unplanned ITAAC Complete Inspection Costs	**($81,600)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
TOTAL NRC UNPLANNED ITAAC COMPLETE INSPECTION COSTS	**($391,680)**

Licensee Early Notification Costs

Cost of Licensee staff time	$100
Hours of Industry staff time	x 1.5
Average Annual No. of ITAAC per Licensee	x 6
Number of Licensees	x 17
Total Annual Licensee Early Notification Costs	**($15,300)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
TOTAL LICENSEE EARLY NOTIFICATION COSTS	**($73,440)**

NRC One Time Costs for Developing Inspection Procedures

Cost of NRC staff time	$120
Hours of NRC staff time	x 80
Total NRC One Time Costs for Developing Inspection Procedures	**($9,600)**

NRC One Time Costs – Developing Guidance

Cost of NRC staff time	$120
Hours of NRC staff time	x 60
Total NRC One Time Costs for Developing Guidance	**($7,200)**

NRC One Time Costs – Developing Processing Infrastructure

Cost of NRC staff time	$120
Hours of NRC staff time	x 48
Total NRC One Time Costs for Develop Processing Infrastructure	**($5,760)**

NRC One Time Costs – Conducting Staff Training

Cost of NRC staff time	$120
Hours of NRC staff time	x 12
Total NRC One Time Costs for Conducting Staff Training	**($1,440)**

Licensee One Time Costs – Developing Processing Procedures

Cost of Industry staff time	$100
Number of Licensees	x 17
Hours of Industry staff time	x 24
Total Industry One Time Costs for Developing Processing Procedures	**($40,800)**

52.99(c)(3)(ii) Supplemental ITAAC Closure Notification

The proposed revisions would require the licensee, after submitting a notification under paragraph (c)(3)(i), to submit a supplemental ITAAC closure notification documenting the resolution of the issue which prompted the paragraph (c)(3)(i) report. The information provided in the notification should be at a level of detail comparable to the ITAAC closure notification under paragraph (c)(1). The dual purposes of the proposed paragraph (c)(3)(ii) notification are (1) to ensure that the NRC has sufficient information, in light of new information developed or identified after the ITAAC closure notification under 10 CFR 52.99(c)(1), to complete all of the activities necessary for the Commission to make a determination on ITAAC, and (2) to ensure that interested persons have access to information on ITAAC at a level of detail sufficient to address the AEA Section 189.a(1)(B) threshold for requesting a hearing.

NRC Costs to Review/Process Supplemental ITAAC

Cost of NRC staff time	$120
Hours of NRC staff time	x 20
Average Annual No. of ITAAC per Licensee	x 6
Number of Licensees	x 17
Total Annual ITAAC Complete Notification Costs	**($244,800)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
TOTAL NRC REVIEW/PROCESS COSTS	**($1,175,040)**

Licensee Costs Supplemental ITAAC Closure

Cost of Licensee staff time	$100
Hours of Industry staff time	x 20
Average Annual No. of ITAAC per Licensee	x 6
Number of Licensees	x 17
Total Annual Supplemental ITAAC Closure Costs	**($204,000)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
TOTAL LICENSEE SUPPLEMETAL ITAAC CLOSURE COSTS	**($979,200)**

NRC One Time Costs – Developing Guidance

Cost of NRC staff time	$120
Hours of NRC staff time	x 120
Total NRC One Time Costs for Developing Guidance	**($14,400)**

NRC One Time Costs – Develop Processing Infrastructure

Cost of NRC staff time	$120
Hours of NRC staff time	x 96
Total NRC One Time Costs for Develop Processing Infrastructure	**($11,520)**

NRC One Time Costs – Conducting Staff Training

Cost of NRC staff time	$120
Hours of NRC staff time	x 24
Total NRC One Time Costs for Conducting Staff Training	**($2,880)**

Licensee One Time Costs – Developing Processing Procedures

Cost of Industry staff time	$100
Number of Licensees	x 17
Hours of Industry staff time	x 48
Total Industry One Time Costs for Developing Processing Procedures	**($81,600)**

52.99(c)(4) ITAAC Closure Documentation

The proposed revisions would require that licensees maintain records of the bases for determining whether a notification of new information on ITAAC closure under 10 CFR 52.99(c)(3)(i) is required and records of the bases for all notifications required under 10 CFR 52.99(c). This would include records supporting initial ITAAC closure letters under paragraph (c)(1), uncomplete ITAAC notifications under paragraph (c)(2), supplemental ITAAC closure letters under (c)(3). The onsite ITAAC closure package would provide the technical basis for the licensee's submittals under 10 CFR 52.99(c). As such, it can be viewed as a "roadmap" documenting how the licensee has established that the activities related to verifying that the ITAAC acceptance criteria are met were accomplished. Licensees would be required to retain these records for a period of 5 years after the date the Commission makes the finding under 10 CFR 52.103(g).

Annual – ITAAC Closure Documentation

Cost of Industry staff time	$100
Hours of Industry staff time	x 1
Number of ITAAC per Lisencee	x 810
Number of Licensees	x 17
Total Annual ITAAC Closure Documentation	**($1,377,000)**
Average Years Needed for Recordkeeping	x 7
Inflationary Ratio to account for 20 year Construction Period	x 1.2
ITAAC CLOSURE DOCUMENTAATION COSTS (Annualized)	**($11,566,800)**

One Time Recordkeeping - ITAAC Closure Documentation

Cost of Industry staff time	$100
Hours of Industry staff time	x 20
No. of ITAAC per Licensee	x 810
Number of Licensees	x 17
ITAAC CLOSURE DOCUMENTATION COSTS (One Time)	**+ ($27,540,000)**
TOTAL ITAAC CLOSURE DOCUMENTATION COSTS	**($39,106,800)**

One Time Costs – Developing Documenting Procedures

Cost of Industry staff time	$100
Number of Licensees	x 17
Hours of Industry staff time	x 80

Total Industry One Time Costs for Developing Documenting Procedures ($136,000)

52.99(f) All ITAAC Complete Notification

The proposed revisions would require licensees to notify the NRC that all ITAAC are complete. At the time the licensee submits the all ITAAC complete notification, the NRC would expect that all activities requiring supplemental ITAAC closure letters have been completed, that the associated ITAAC determination bases have been updated, and that all required notifications under paragraphs (c)(3) have been made.

Licensee All ITAAC Complete Notification

Cost of Industry staff time	$100
Hours of Industry staff time	x 8
Average Annual No. of ITAAC per Licensee	x 6
Number of Licensees	x 17
Total Annual All ITAAC Complete Notification Costs	**($81,600)**
Years to Complete Plant Construction	x 4
Inflationary Ratio to account for 20 year Construction Period	x 1.2
ALL ITAAC COMPLETE NOTIFICATION COSTS	**($391,680)**

Licensee One Time Costs – Developing Processing Procedures

Cost of Industry staff time	$100
Number of Licensees	x 17
Hours of Industry staff time	x 8
Total Industry One Time Costs for Developing Processing Procedures	**($13,600)**

NRC One Time Costs – Developing Guidance

Cost of NRC staff time	$120
Hours of NRC staff time	x 20
Total NRC One Time Costs for Developing Guidance	**($2,400)**

NRC One Time Costs – Develop Processing Infrastructure

Cost of NRC staff time	$120
Hours of NRC staff time	x 16
Total NRC One Time Costs for Develop Processing Infrastructure	**($1,920)**

NRC One Time Costs – Conducting Staff Training

Cost of NRC staff time	$120
Hours of NRC staff time	x 4
Total NRC One Time Costs for Conducting Staff Training	**($480)**