POLICY ISSUE
(Information)

August 14, 2009 SECY-09-0113

FOR: The Commissioners

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

SUBJECT: UPDATE ON THE DEVELOPMENT OF CONSTRUCTION ASSESSMENT PROCESS POLICY OPTIONS AND THE CONSTRUCTION INSPECTION PROGRAM INFORMATION MANAGEMENT SYSTEM

PURPOSE:

In response to Staff Requirements Memorandum (SRM) M081022, "Staff Requirements - Periodic Briefing on New Reactor Issues," dated December 5, 2008, this paper updates the Commission on the progress by the U.S. Nuclear Regulatory Commission (NRC) staff towards the development of the construction assessment program policy options. This paper also provides an update regarding the development and embodiment of safety culture for the construction inspection program (CIP), including inputs and thresholds for the construction response table (CRT). In addition, this paper updates the Commission on the staff's progress in developing the software enhancements to the Construction Inspection Program Information Management System (CIPIMS).

BACKGROUND:

In the SRM associated with SECY-07-0047, "Staff Approach to Verifying the Closure of Inspections, Tests, Analyses, and Acceptance Criteria Through a Sample-based Inspection Program," dated May 16, 2007, the Commission directed the staff to engage industry and interest group stakeholders to obtain their views on the CIP using a public meeting approach similar to that used during the development of the Reactor Oversight Process (ROP). The Division of Construction Inspection and Operational Programs in the Office of New Reactors

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(NRO) and the Region II Center for Construction Inspection have conducted numerous public meetings with the industry and other NRC stakeholders for the past 2 years to discuss the development of programs and procedures related to new reactor construction under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." One of the primary purposes of these meetings has been to obtain stakeholder input and feedback for the construction assessment program.

The staff considered stakeholder input received during these public meetings to develop the construction assessment program, which is described in Inspection Manual Chapter (IMC) 2505, "Periodic Assessment of Construction Inspection Program Results," dated October 20, 2008. The staff submitted SECY-08-0155, "Update on the Development of the Construction Inspection Program for New Reactor Construction under 10 CFR Part 52," dated October 17, 2008, and subsequently briefed the Commission on October 22, 2008. Following the briefing, the Commission directed the staff in SRM M081022 to reconsider the construction assessment process as presented in IMC 2505 and propose policy options to the Commission. The Commission further directed that the staff proposal should address the inclusion in the construction oversight process of objective elements such as construction program performance indicators (PIs) and significance determination processes (SDPs) analogous to those used in the ROP.

DISCUSSION:

The staff has reconsidered the construction assessment process by engaging stakeholders in a series of meetings intended to identify alternative means of assessing licensee performance to those described in SECY-08-0155 and IMC 2505, and to develop construction assessment program options for Commission consideration. The staff re-reviewed ROP basis documents in an effort to identify additional features that could be applied in the development of the construction assessment process.

The staff presented assessment program concepts, including options to develop and implement construction PIs and SDPs, to external stakeholders during numerous public workshops and meetings. The staff considered the feedback received from these meetings with stakeholders in further development of the options. The staff also considered feedback contained in a December 5, 2008, Nuclear Energy Institute (NEI) letter to the Commission entitled "NRC Oversight of Construction Activities" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092080426). Specifically, NEI did not support the development and use of PIs and SDPs for construction.

As the staff was concluding its work to develop construction assessment program options through public workshops, the industry developed an independent proposal for the construction assessment process. On July 2, 2009, NEI submitted a letter to the NRC, entitled "Proposed Construction Inspection Assessment Process" (ADAMS Accession No. ML091831352). This effort, led and coordinated by NEI, presents a concept that is similar in structure to the ROP regulatory framework. This concept does not include the use of PIs and proposes that the significance of findings be determined using a method other than traditional enforcement. The concept lacks detail and requires developmental work.

The staff has considered NEI's framework and supporting text. In light of the timing and content of NEI's proposal, the staff believes that further collaboration with all stakeholders through public workshops is necessary to determine the most comprehensive set of construction assessment program policy options. NRC consideration and further development of NEI's recent proposal will necessitate the allocation of additional staff resources to provide the Commission with policy options.

During these meetings, the staff plans to thoroughly explore stakeholder proposals and to develop policy options for Commission consideration. With this in mind, the staff plans to provide construction assessment program policy options to the Commission in November 2010. This schedule will support the development of policy options prior to the first unit combined operating license and the full onset of construction oversight activities. To support the first limited work authorization, staff will revise IMC 2505 and IMC 0613 "Documenting 10 CFR 52 Construction and Test Programs" to incorporate a safety culture approach, such that a construction assessment program can be implemented on an interim basis.

As the staff considers the NEI proposal, it will build on the extensive work already completed in developing policy options. The enclosures to this document describe the results of work conducted to date by the staff in developing alternatives for the construction assessment program. Enclosure 1 describes the features that the staff believes the construction assessment program policy options should include, such as periodic assessments of licensee construction performance and areas important to safety culture, trending and reacting to issues to prevent larger issues from developing, a CRT that provides for a range of NRC actions commensurate with the significance of issues identified during the CIP, and routine and clear communication of assessment results to stakeholders. Consistent with the ROP, traditional enforcement should be used to disposition issues that have an impact on the regulatory process, are willful, or have actual safety consequences.

Enclosure 2 details the methods the staff has considered, to date, to evaluate the significance of findings. Enclosure 3 describes staff efforts to develop construction PIs. The NRC could implement traditional enforcement processes to assess CIP findings. While traditional enforcement could be readily used, the development of a SDP is potentially challenging and will require additional work. Alternatively, the staff could develop a construction SDP to assess the significance of CIP findings. The establishment of a risk-informed threshold for construction PIs has proven to be particularly challenging because of the lack of performance data to benchmark new construction PI thresholds.

The staff and our stakeholders recognize that a strong safety culture during new reactor construction is important to ensuring that the newly constructed plant is in compliance with the design and will be capable of operating safely following construction. In order to remain aligned with agency safety culture developments, NRO representatives have participated in NRC staff efforts pursuant to the Commission direction provided in COMSECY COMGBJ-08-0001, "A Commission Policy Statement on Safety Culture," dated February 25, 2008. NRO will continue to participate in these NRC staff activities and will develop a long-term approach to safety culture that is consistent with that developed for the ROP.

The staff recognizes the need to assess areas important to safety culture at the beginning of NRC inspections associated with licensed construction activities. The staff reviewed domestic and international construction events and evaluated the applicability of the existing 13 safety culture components to these events. As a result of this review, it was determined that all 13 of these components were integral to a comprehensive NRC oversight of safety culture as applied to new reactor construction activities. Therefore, the staff intends to implement a near-term approach to safety culture which closely resembles that already being implemented in the operating reactor assessment program. This oversight role includes documenting cross-cutting aspects during the course of NRC inspections, evaluating these findings against a pre-defined set of criteria to determine if a significant concern exists, and conducting appropriate follow-up actions applied using a graded approach. Significant concerns will be treated in a manner analogous to the ROP's substantive cross-cutting issues.

On July 23, 2009, the staff conducted a public meeting with stakeholders and informed them of the planned near-term approach to incorporate safety culture into construction oversight, in a manner that closely resembles the ROP. The staff also discussed long-term options under consideration for incorporating safety culture into the construction assessment process. The staff has three public meetings planned for the remainder of calendar year 2009 and has tentatively scheduled seven public meetings for 2010.

The staff has also continued development of CIPIMS, which will be used for documenting results from 10 CFR Part 52 construction inspections. CIPIMS is currently available as part of the Reactor Programs System, but work is in progress to migrate CIPIMS to the Enterprise Project Management (EPM) platform. The move to the EPM environment will provide inspectors with tools such as spell check and rich text editing. This transition is expected to be completed by the end of fiscal year (FY) 2009. Continued enhancements will be made to CIPIMS as experience with its use and capabilities is gained.

COMMITMENTS:

The staff plans to issue revisions to IMC 2505 and IMC 0613 by November 30, 2009, including an approach to safety culture similar to the existing ROP, so that a complete assessment process can be implemented on an interim basis to support the issuance of the first limited work authorization.

The staff intends to continue working with industry and other stakeholders on the development of assessment program policy options, including long-term options that are being considered for incorporating safety culture into the construction assessment process. The staff plans to submit construction assessment program policy options to the Commission by November 2010.

RESOURCES:

The staff estimates that 3 FTE in FY 2010 and 1 FTE in FY 2011 will be required to complete the construction assessment program policy options. The staff intends to redirect FTE from other lower priority program development work. NRO will work with the Region II Center for Construction Inspection to ensure the redirected FTE will not impact timely implementation of the construction inspection program.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections.

/RA/

Michael R. Johnson, Director
Office of New Reactors

Enclosures:
1. Construction Assessment Program Attributes
2. Evaluation of CIP Findings
3. PI Considerations

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Construction Assessment Program Attributes

This enclosure provides background information on the development of the construction inspection program in light of lessons learned from NUREG-1055, “Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants: A Report to Congress,” and changes to the construction licensing framework as reflected in Title 10 of the *Code of Federal Regulations* (10 CFR) 52. Additionally, this enclosure details the construction assessment program as outlined in IMC 2505, “Periodic Assessment of Construction Inspection Program Results” including the near-term safety culture approach that closely resembles that being implemented for operating reactors as well as a longer term approach.

<u>Background</u>

NUREG-1055 was issued May 1984 and detailed lessons learned during the early days of construction under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”. This report concluded that the U.S. Nuclear Regulatory Commission (NRC) was slow to detect and take strong action on significant quality problems that developed during nuclear power plant construction projects. In addition, the NRC did not have a formal assessment process in place to evaluate the performance of construction permit holders. Following the accident at Three Mile Island, the NRC initiated an effort to better address licensee performance through the Systematic Assessment of Licensee Performance (SALP) program. Under the SALP program, the NRC periodically reviewed the overall performance of each nuclear power plant licensee (both construction permit holders and operating license holders) in a number of different functional areas. Each functional area evaluated was assigned to one of three categories to indicate whether more, less, or about the same level of NRC inspection and licensee attention was appropriate for the coming period. The SALP assessment was intended to be sufficiently diagnostic to provide a rational basis for assessing licensee performance, allocating NRC inspection resources, and providing meaningful guidance to licensee management.

In 1991, the NRC began work to revise the construction inspection program (CIP) to address programmatic weaknesses that had been identified during the inspection and licensing of plants in the 1980s. This project was suspended in late 1994 because of the lack of nuclear power plant construction activities. Before that project was suspended, the staff had worked to document the lessons learned from previous NRC construction inspections and from reviews of inspection practices overseas and modular construction techniques used in the U.S. shipbuilding industry. During this period, the staff continued to use the SALP program to assess licensee performance. In 1998, the NRC suspended the SALP program and, for operating reactors, eventually replaced it with the Reactor Oversight Process (ROP) in 2000.

The NRC renewed work to revise the CIP in 2001 when it formed the Construction Inspection Team, composed of representatives from each region, new reactor licensing staff, and inspection program management, and tasked it with updating the inspection and assessment

program for use in inspecting reactors to be licensed and constructed under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”
The current effort to develop the CIP has focused on ensuring that the inspection program collects the information to support the Commission determination under 10 CFR 52.103(g), on whether the acceptance criteria in the combined license are met. It also focuses on addressing the various lessons learned from the NRC’s previous construction inspection experience (e.g., trend evaluations and early identification of problems and other lessons learned as detailed in NUREG-1055, “Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants: A Report to Congress,” issued May 1984).

Construction Assessment Process

The construction assessment process will begin after the NRC issues a limited work authorization and/or a combined license; the NRC has implemented either Inspection Manual Chapter (IMC) 2503, “Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” or IMC 2504, “Construction Inspection Program—Non-ITAAC Inspections,” both dated October 3, 2007; and there has been sufficient activity occurring for any assessment to be meaningful. The construction assessment process will continue until oversight transitions to the ROP. The agency will base the assessment period on the anniversary date on which the construction assessment process began. Once a plant transitions to the ROP, the assessment process will revert to the calendar year process outlined in IMC 0305, “Operating Reactor Assessment Program,” dated April 9, 2009.

The staff plans to implement a construction response table (CRT) analogous to the ROP action matrix. The staff developed the CRT according to the philosophy that, within a certain level of performance (i.e., the Baseline Program column), licensees would address their performance issues without additional NRC engagement beyond the baseline inspection program. Agency action beyond the baseline inspection program will normally occur only if assessment input thresholds are exceeded. The CRT identifies the range of NRC and licensee actions and the appropriate level of communication for varying levels of performance.

The overall response to licensee performance will be determined by the inputs specified in the option approved for implementation by the Commission. The NRC will use a graded approach to determine the response to the identified issues. This graded approach will result in an increase in sampling in the area(s) of concern; an increase in the inspections, tests, analyses, and acceptance criteria (ITAAC) being inspected; and/or the issuance of a confirmatory action letter, demand for information, and/or order. Increased inspection, whether increased sampling or the selection of additional ITAAC, will be conducted through the use of expanded inspections.

The construction assessment process includes three basic parts: continuous assessment, quarterly assessment, and semiannual performance review (SPR) assessment. The site construction team will continuously assess licensee performance by evaluating violations as they are identified and adjust the inspection plan as necessary. Additionally, the staff will conduct assessments on a quarterly basis and take appropriate actions in accordance with the CRT. The licensee and the public will be informed of the results of the Agency’s assessment on a semi-annual basis. Additionally, the Agency will hold an annual public meeting in the vicinity of the construction site to communicate performance assessment results.

Regardless of which assessment option is approved, consistent with the Enforcement Policy, several types of violations will continue to require the use of traditional enforcement, including the possible issuance of fines. Examples include the following:

- discrimination against workers for raising safety issues or other willful violations

- actions that may adversely affect the NRC's ability to monitor utility activities, including failure to report required information, failure to maintain accurate records, or failure to provide the NRC with complete and accurate information

- incidents with actual safety consequences

Safety Culture Approach

In order to remain aligned with agency safety culture developments, NRO representatives have been appropriately involved with NRC staff activities pursuant to the Commission direction provided in COMSECY COMGBJ-08-0001, "A Commission Policy Statement on Safety Culture," dated February 25, 2008. The formulation of a long-term agency level approach to safety culture is ongoing. Much work remains to be accomplished on an agency level in the development of a long-term approach to safety culture. The staff recognizes the need to assess areas important to safety culture at the beginning of NRC inspections associated with licensed construction activities. Therefore, the staff intends to implement a near-term safety culture approach which closely resembles that already being implemented in the operating reactor assessment program. This oversight role includes documenting cross-cutting aspects during the course of NRC inspections, evaluating these findings against a pre-defined set of criteria to determine if a significant concern exists, and conducting appropriate follow-up actions applied using a graded approach.

The staff determined that all 13 of the safety culture components were integral to a comprehensive NRC oversight of safety culture as applied to new reactor construction activities and would reveal themselves in an observable manner. Similar to the ROP, the staff plans to evaluate inspection findings against nine of the components in the baseline inspection program and against all 13 components in the supplemental inspection program. While the 13 safety culture components are applicable to construction environments, some of the aspects associated with each of the components are not applicable. The staff, with stakeholder input, is developing construction program safety culture aspects for each of the 13 components.

Similar to the operating reactor assessment program, during periodic assessments of licensee performance, the staff plans to analyze CIP findings to determine if a theme existed for violations identified during the assessment period (analogous to the ROP's substantive cross-cutting issues). Themes for violations would be referred to as construction safety focus issues. To remain consistent with the ROP, the staff plans to remove construction safety focus issues as an input to the CRT. Thresholds for the identification of construction safety focus issues are being developed by the staff.

The staff will modify IMC 2505, "Periodic Assessment of Construction Inspection Program Results," and IMC 0613, "Documenting 10 CFR Part 52 Construction and Test Inspections," both dated October 20, 2008, to incorporate the safety culture approach outlined above. The staff expects to issue revisions to IMCs 2505 and 0613 by November 30, 2009.

NRO will continue to participate in NRC staff efforts and will develop a long-term approach to safety culture that is consistent with the approach developed for the ROP. In accordance with the Commission's direction in Staff Requirements Memorandum M081022, "Staff Requirements - Periodic Briefing on New Reactor Issues," dated December 5, 2008, the NRO staff will continue to work with industry and other stakeholders on the development and embodiment of safety culture within the CIP, and it will keep the Commission informed as the approach to safety culture evolves.

Evaluation of Construction Inspection Program Findings

Adequate protection of public health and safety and the assurance of the common defense and security are fundamental regulatory objectives. Licensee compliance with U.S. Nuclear Regulatory Commission (NRC) requirements plays an important role in giving the NRC confidence that safety is being maintained, in that adequate protection is predicated on compliance with NRC requirements. When a licensee does not comply with NRC requirements, the NRC implements its Enforcement Policy. Once the agency identifies a noncompliance, the staff assesses its significance or severity. The significance of a construction inspection program (CIP) violation can be determined through several different methods. This enclosure describes the various methods the staff considered for use in determining the significance of CIP violations.

Traditional Enforcement

The staff of the Office of New Reactors (NRO) has worked closely with the Office of Enforcement in developing a revision to the Enforcement Policy to address circumstances associated with new reactor construction under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff, with stakeholder input, has developed draft examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations identified through the CIP, including violations that are associated with the inspections, tests, analyses, and acceptance criteria (ITAAC) closure and maintenance process. The staff has issued a proposed revision to the Enforcement Policy, including the examples of violations, for public comment. The staff is in the process of incorporating these comments and will forward the proposed policy to the Commission for approval. Using examples as guidance in dispositioning findings should provide transparency and predictability to the oversight of licensee performance.

The CIP will review the adequacy of the development and implementation of construction and operational programs. Traditional enforcement is well suited to disposition findings associated with inadequate program development and implementation. The inclusion in 10 CFR Part 52 of inspections, tests and analyses with specific acceptance criteria as requirements presents objective performance criteria that do not exist in the Reactor Oversight Process (ROP) or in previous construction projects licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Thus, findings associated with ITAAC closure and maintenance can be objectively dispositioned using traditional enforcement.

As discussed in Enclosure 1, regardless of which assessment option is approved, the following violations will continue to require the use of traditional enforcement, including the possible issuance of fines. Examples include the following:

- discrimination against workers for raising safety issues or other willful violations

- actions that may adversely affect the NRC's ability to monitor utility activities, including failure to report required information, failure to maintain accurate records, or failure to provide the NRC with complete and accurate information

- incidents with actual safety consequences

Significance Determination Process

In the ROP, the staff developed a method for assigning a qualitative or quantitative probabilistic risk characterization to inspection findings related to reactor safety. This risk characterization tool was the first of a set of tools that became central elements of the significance determination process (SDP) to determine the significance of reactor inspection findings consistent with the thresholds used for plant performance indicators. This allowed the staff to use both inspection findings and performance indicators consistently as inputs to the plant performance assessment portion of the ROP. Subsequently, the staff developed other SDP tools to characterize the safety significance of issues associated with emergency preparedness, occupational and public radiation safety, physical protection, fire protection, shutdown operations, containment integrity, operator requalification, and steam generator tube integrity. These SDP tools used qualitative or quantitative risk evaluation methods when possible. SDPs that could not be related to core damage or containment failure risk used other rationale for assigning significance. Historically, such other factors included those listed in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, issued November 2002, such as maintaining defense in depth, compliance with regulations, engineered safety margins, and expert staff judgment.

Given that fuel will not be loaded until construction is complete, core damage is not a direct consideration and findings identified during plant construction have no actual radiological public health and safety risk at that time. It is possible to project the risk significance finding as if the findings were to remain undiscovered until after plant operations began. However, NRC practice has been to determine the risk of findings based on the conditions that existed when the deficiency occurred. For instance, the ROP only uses full-power SDPs when a plant is above residual heat removal system entry conditions. The staff has developed a shutdown operation SDP for use when the plant meets residual heat removal entry conditions.

An additional challenge in attempting to project the risk of findings into the period of plant operations would be the large uncertainty introduced into the process when considering the frequency of the initiating event and operator actions that could be taken to mitigate the condition. With a large uncertainty, an SDP result is less accurate and provides less value to the assessment process. Nonetheless, the staff considered methods such as a risk matrix, a construction experience risk determination model, and reliability growth models to project the risk of construction findings into plant operations. It is worth noting that the ITAAC prioritization methodology provided a risk-informed selection of ITAAC for inspection. The inclusion of ITAAC in combined licenses provides objective measures that the staff can use to determine the quality of construction activities.

Given current risk practices and the large uncertainty involved with projecting the risk of construction findings into the period of operations, the staff believes that if a construction SDP is developed, it is most practical to develop one that assesses the effect of findings on plant construction. Such an SDP would be deterministic in nature and would be structured similarly to radiation protection, emergency preparedness, security, and operator licensing SDPs developed and used in the ROP. The construction SDP goals should be based on successful completion of ITAAC.

Risk Matrix

The SDP for inspection findings of degraded performance of structures, systems, and components (SSCs) in operating reactors uses as input the estimated impact on core damage frequency and/or large early release frequency, along with the estimated duration of the degraded condition. This formulation is highly quantitative in nature and reasonably objective given the inputs that are assumed. For new reactor construction, it is not possible to replicate these elements. However, the concept of a two-dimensional risk matrix that includes a measure of the risk importance of the SSC in question, along with the degree of nonconformance, could provide elements of a somewhat objective risk-informed and performance-based construction oversight program.

One dimension would be the risk importance of the SSC in question. In the Maintenance Rule (10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”), as well as in 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” probabilistic risk assessment importance measures such as Fussell-Vesely and risk achievement worth take on special meaning. These risk importance measures are available for all the new standardized reactor designs. Either these values or the ITAAC ranking of the SSC could be used to determine the risk importance of the SSC in question. A three-group categorization of risk importance (low, high, very high) lends itself to deriving a four-tier finding significance classification (green, white, yellow, and red). The three-group categorization could be increased to four groups to provide a more expanded scale for risk importance.

The second dimension of the matrix would be some measure of the degree of nonconformance. The details of this element could involve such attributes as the following:

- the severity of the nonconformance
- repetition of the nonconformance
- the number of opportunities for self-discovery that were missed
- the duration of the nonconformance

The staff would evaluate the findings and assign them a color corresponding with placement in the matrix.

Construction Experience Risk Determination Model

NRO Office Instruction NRO-REG-112, "New Reactor Construction Experience," dated March 31, 2009, includes a methodology for screening construction or operating events for safety significance and applicability to new reactor designs. The results of this determination are used to identify appropriate NRC followup actions, including trending. The screening process measures the safety significance of construction or operational events, because of design or construction errors, based on two main factors: (1) the degradation of barriers (i.e., reduction in defense in depth), and (2) the likelihood that the failure would not be detected before operation or the period of time it remained undetected during operation. Additionally, the staff considers other factors that may increase the safety significance level on a scale of 1 to 5. These additional factors include common-cause failures, programmatic deficiencies, and occurrences of similar events. For screening purposes, all defects are assumed to stay hidden until after operations begin.

The staff considered this approach as a possible model for an "SDP-like" process for new reactors. A key weakness in this approach is that the process requires a deficiency to be discovered during operations. Therefore, this approach would not be applicable to the first built plant and would be of very limited use for the next few plants. Another challenge is the number of deficiencies that the staff could not process through this model. Although this approach would apply to hardware deficiencies, it is not applicable to programmatic deficiencies.

A possible use of this process may be as an input to a potential performance indicator that measures appropriate licensee response to NRC generic communications, if the agency issued one as result of the deficiency. Another possible use may be as a collection or database of information that could be used as a basis for a longer term project for developing an "SDP-like" process.

Performance Indicator Considerations

This enclosure provides a discussion of staff activities and considerations in developing performance indictors for construction inspection. The objective of the Reactor Oversight Process (ROP) for operating reactors is to monitor performance in three broad strategic areas: reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety for both plant workers and the public during routine operations, and protection of the plant against sabotage or other security threats. To measure plant performance, the operating reactor oversight program focuses on seven specific cornerstones that support the safety of plant operations in the three broad strategic areas.

For each cornerstone, the U.S. Nuclear Regulatory Commission (NRC) develops findings from inspections and licensees collect performance indicator (PI) data. The staff evaluates inspection findings for safety significance using a significance determination process and compares PI data against prescribed thresholds. The staff then assesses the resulting information and determines an appropriate NRC response using the guidelines in an action matrix, which typically includes supplemental inspections for selected issues. The agency takes enforcement action on significant inspection findings, as appropriate.

The NRC developed the ROP with the benefit of four decades of operational experience and, generally speaking, steadily improving plant performance. Before implementing the ROP, industry (Nuclear Management and Resources Council/Nuclear Energy Institute/Institute of Nuclear Power Operations) and the NRC (Office for Analysis and Evaluation of Operational Data, Office of Nuclear Regulatory Research) maintained plant operational PIs that were used in various ways to assess overall industry and individual plant performance. These PIs were used as a basis to develop the PIs that were implemented as part of the ROP assessment process. PIs are a means of obtaining information related to the performance of certain key attributes in each of the cornerstone areas. They indicate problems that, if uncorrected, may increase the probability and/or the consequences of an off-normal event. The staff uses data submitted by each licensee to calculate PI values and then compares these values to objective thresholds to determine the performance band associated with the values.

The NRC developed the significance determination process to help inspectors determine the safety significance of inspection findings. In developing the operating reactor performance assessment process, one of the tasks was to establish thresholds for PIs and corresponding thresholds for inspection findings, so that indications of performance degradation obtained from inspection findings and from changes in PI values could be put on an equal footing. The concept for setting these performance thresholds included the consideration of risk and the regulatory response to different levels of licensee performance. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” served as the basis for establishing these performance thresholds and adopted core damage frequency and large early release frequency as the metrics for the characterization of risk.

The nuclear power construction industry has been dormant and, as a result, is much less mature than the operating reactor industry. While it is possible to develop PIs for construction performance, there will be no history for the PIs, and establishing appropriate thresholds will be very challenging without an accumulated history of data.

In response to Commission direction in Staff Requirements Memorandum M081022, "Staff Requirements - Periodic Briefing on New Reactor Issues," dated December 5, 2008, the staff attempted to identify viable construction PIs. The staff first analyzed ROP PIs to determine if any could be applicable to a construction environment. Table 1 includes the results of this analysis. The staff concluded that it may be possible to implement a modified version of the PI for occupational exposure control effectiveness. No other ROP PIs appeared to apply in a construction environment.

The staff then attempted to identify viable construction PIs. The staff used the following basis, the same as that used for initial ROP PI development:

- capable of being measured objectively
- thresholds to guide NRC and licensee actions
- reasonable sample of performance
- valid and verifiable indication of performance in the area being measured
- encouragement of appropriate licensee and NRC actions
- provision of sufficient time for the NRC and the licensee to correct performance deficiencies before the deficiency poses an undue risk to public health and safety

The staff identified approximately 60 potential construction PIs and binned them into three categories, with Category A having the most likelihood of development. PIs in Category B may be implemented with fundamental changes, and those in Category C were not likely to be implemented. Generally, an answer of "no" to three or more of the above criteria resulted in the assignment of a PI to Category C; an answer of "no" to two criteria resulted in the assignment of a PI to Category B; and an answer of "no" to one criterion resulted in the assignment of the PI to Category A. Of the approximately 60 potential construction PIs considered by the staff, 13 were determined to be either Category A or B. The staff sought stakeholder feedback regarding possible PIs in several public meetings and, after further deliberations, determined that it may be possible to develop the following PIs:

- failure to adequately assess or respond to generic communications
- overdue pending design changes
- overdue safety-significant corrective action program issues
- due date extensions to safety-significant corrective action program issues

- ineffective corrective actions
- number of re-opened inspections, tests, analyses, and acceptance criteria (ITAAC)
- number of errors resulting from inadequate training
- extent of condition review
- ratio of NRC-identified findings to licensee-identified findings
- occupational exposure control

During development of the ROP PIs, the staff conducted extensive benchmarking activities. The NRC used 17 plants as case studies to validate the selected equipment PI and thresholds. It performed similar benchmarking for emergency planning, radiological effluents, and occupational exposure controls. To determine the green/white threshold (for operational PIs), the staff needed to determine the acceptable level of performance. It applied a statistical approach for plants with an acceptable performance (a generically achievable level of performance) while developing the threshold. For drill and exercise performance, it determined the white band by using two standard deviations below the average, and the yellow band by using three standard deviations below the average. The evaluation was a measure of the successful classifications/protective action recommendations divided by the total opportunities. The following seven questions represent the extent to which the staff examined the overall impact of the ROP PIs:

1) Do the PIs differentiate among superior, average, declining, and “watch list” plants, as determined at the Senior Management Meeting (SMM)?

2) How effective are individual PIs at differentiating among plants with different levels of performance (from the SMM)?

3) Do the PIs demonstrate timely response?

4) Do the PIs show declining trends for plants identified as declining by the SMM?

5) Do the PIs show declining trends before accident sequence precursor events?

6) How well does the set of PIs conform to those selected by Arthur Andersen for use in the trending methodology that was used in the SMM process?

7) Do small decreases in the green/white thresholds capture more of the watch list and declining plants?

The establishment of a risk-informed threshold for construction PIs has proven to be particularly challenging. As mentioned previously, the staff initially developed the ROP PI thresholds based on years of historical licensee performance data collected by the NRC and industry. The staff has been unable to identify similar data for plants under construction. No performance data exists against which to benchmark new construction PI thresholds. The applicable regulatory requirements for construction activities call for the licensee to demonstrate that all ITAAC are met. The staff will continue to explore PIs focused on the licensee's successful completion of ITAAC.

The staff understands that utilities planning new construction projects intend to develop and implement PIs to monitor construction activities. The staff plans to monitor industry's use of PIs and will incorporate the experience gained during these activities as directed by the Commission.

Table 1 Comparison of Existing PIs for the ROP to the New Reactor Construction Environment

PI	Description	New Construction Environment
(1) Unplanned SCRAMS per 7,000 Critical Hours	Monitors the number of unplanned SCRAMS—measures Human Error, Procedure Quality, Design and Equipment Performance	No critical operations
(2) Unplanned SCRAMS with Complications	Subset of unplanned SCRAMS while critical that require additional operator actions—measures Human Error, Procedure Quality, Design and Equipment Performance	No power operations
(3) Unplanned Power Changes per 7,000 Critical Hours	Monitors unplanned power changes (excluding SCRAMS) potential to challenge safety functions, considered to be a leading indicator of declining performance – measures Human Error, Procedure Quality, Design, and Equipment Performance	No power operations
(4) Mitigating System Performance Index	Monitors readiness of important safety systems – measures Configuration Control, Equipment Performance, and Human Performance	Systems will be placed in isolation or standby once ITAAC are completed, no operations in a condition when the safety-significant equipment is required to be available; no potential to affect CDF.
(5) Safety System Functional Failures	Equipment failures that could have prevented: (a) Ability to shutdown or maintain reactor shutdown (b) Remove residual heat (c) Control release of radioactive material (d) Mitigate the consequence of an accident	No direct comparison; new definition can be a measure of ITAAC preservation for safety-significant systems
(6) Reactor Coolant System (RCS) Specific Activity	Monitors the integrity of fuel cladding – measures Design Control, Configuration Control, Cladding Performance, Procedure Quality, and Human Performance	No power operations, no source term production, no challenge to fuel
(7) Reactor Coolant Leakage	Monitors integrity of RCS pressure boundary – measures RCS equipment and Barrier Performance	Only relevant once RCS boundary is established, construction cycle not impacted.

(8) Drill/Exercise Performance	Timely and accurate licensee performance in drills and exercises (notifications and protective action recommendation)—measures facilities and equipment, procedure quality, and emergency response organization (ERO)	Only one drill required before fuel load
(9) ERO Drill Participation	Percentage of key ERO members – measures facilities and equipment, procedure quality, and ERO performance	Little demand for requalification before transition to ROP
(10) Alert and Notification System (ANS) Reliability	Monitors reliability of offsite ANS, link to public for implementation of protective action recommendation - measures facilities and equipment	Not required before fuel load, not relevant during construction
(11) Occupational Exposure Control Effectiveness	Control of access to and work activities within radiologically significant areas of the plant – measures plant facilities/equipment and instrumentation, program/process, and human performance	Radiological sources limited to radiography before reactor startup
(12) RETS/ODCM Radiological Effluent Occurrence	Assesses the performance of radiological effluent control program – measures plant facilities/equipment and instrumentation, program/process, and human performance	No radiological effluents before fuel load
(13) Protected Area Equipment	Index that compares the amount of time closed circuit television cameras and intrusion detection system are unavailable, as measured by compensatory hours, to the total hours in the period	No current security requirements for construction