POLICY ISSUE
INFORMATION

May 24, 2006 SECY-06-0122

FOR: The Commissioners

FROM: Luis A. Reyes
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SUBJECT: SAFETY CULTURE INITIATIVE ACTIVITIES TO ENHANCE THE REACTOR OVERSIGHT PROCESS AND OUTCOMES OF THE INITIATIVES

PURPOSE:

The purpose of this paper is to provide the Commission with information on the staff's activities, current status, and plans to enhance the Reactor Oversight Process (ROP) to more fully address safety culture, in response to Commission direction in Staff Requirements Memoranda (SRMs) on SECY-04-0111, "Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture" and SECY-05-0187, "Status of Safety Culture Initiatives and Schedule for Near Term Deliverables."

SUMMARY:

Based on Commission direction, the staff developed an approach with the participation of stakeholders to enhance the ROP to more fully address safety culture. The enhancements to the ROP include better aligning the three cross-cutting areas to those aspects of performance that are important to safety culture. The staff also adjusted selected baseline, event response, and supplemental inspection procedures and inspection manual chapters. These adjustments are within the framework of the ROP and are consistent with the principles that guided the development of the ROP. The staff developed inspector training on safety culture and the

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changes to the ROP. The staff has provided computer-based training and is conducting training at regional counterpart meetings. In the longer term, the staff will work with the Technical Training Center (TTC) to incorporate aspects of the safety culture initiative into initial training for new inspectors and continuing training for existing inspectors. The modified ROP will be implemented by July 1, 2006. The staff will assess the first 18 months of implementation of the safety culture enhancements as part of the annual ROP assessment and will make changes as needed. The staff will update the Commission on the status of this effort, including the results of the implementation assessment, in the Commission Paper on the Reactor Oversight Process Self-Assessment for Calendar Year 2007.

BACKGROUND:

On July 1, 2004, the staff developed SECY-04-0111 to provide options for Commission consideration relative to a safety conscious work environment and safety culture in response to lessons learned from the reactor vessel head degradation event at Davis-Besse. The staff undertook activities to enhance the ROP to more fully address safety culture based on the Commission's direction in SRM/SECY-04-0111, dated August 30, 2004.

The staff provided a status of those activities in SECY-05-0187, dated October 19, 2005. The paper addressed staff's activities up to that point which included issuing Regulatory Issues Summary (RIS) 2005-18, "Establishing and Maintaining a Safety Conscious Work Environment," dated August 25, 2005; establishing a Safety Culture Steering Committee, Safety Culture Working Group, and Support Team to identify and implement activities in response to the SRM; establishing a safety culture Web page; and engaging external stakeholders to obtain their input on safety culture initiative activities. The staff also described the proposed schedule to complete development of the changes to the ROP, provide training to the inspectors on safety culture and changes to the ROP, and begin implementation of the modified ROP. SECY-05-0187 included an enclosure which described the changes associated with safety culture issues made to the ROP following the Davis-Besse Lessons Learned Task Force report. The Commission provided direction to the staff in SRM/SECY-05-0187, dated December 21, 2005. The Commission's major direction to the staff in SRM/SECY-04-0111 and SRM/SECY-05-0187 is provided in Enclosure 1.

DISCUSSION:

Provided below is a description of stakeholder involvement in the development of the approach to enhance the ROP, development of the approach and key features, current status, and remaining key activities and schedule.

Stakeholder Involvement in the Development of an Approach to Enhance the ROP to More Fully Address Safety Culture

Based on verbal feedback from the Commission in early November 2005, the staff engaged external stakeholders to revisit and revise the planned approach for enhancing the ROP that was described in SECY-05-0187. In developing the approach, the staff was guided by the regulatory principles of the ROP, so that any proposed changes to oversight activities and outcomes remain transparent, understandable, objective, predictable, risk-informed, and performance-based. To ensure that appropriate inspection expertise would inform the safety culture initiative, the safety culture working group was expanded to include a dedicated representative from each of the regions. The regional team was comprised of two regional Branch Chiefs, a senior resident inspector, and a senior project engineer.

Throughout the period November 2005 through February 2006, the staff with the close involvement of external stakeholders developed an approach to modify the ROP to more fully address safety culture. This was accomplished, in part, through frequent public meetings. The staff provided all public meeting presentations and other meeting materials on the safety culture Web page. Stakeholders who were unable to attend the meetings onsite participated through telephone conferences. The staff provided the Inspection Procedures (IPs) and Inspection Manual Chapters (IMCs) on the Web page and has received comments from the external stakeholders. The staff plans to place the resolution of the comments on the Web page within the next few weeks.

With regard to other stakeholder interactions, the staff sought to keep internal stakeholders involved throughout the development process and to provide an opportunity for feedback on the safety culture initiative activities/outcomes. This was accomplished through activities such as regular meetings of the Safety Culture Steering Committee, video conferences with regional management and program office management, briefings for the Executive Director for Operations and the applicable Deputy Executive Directors, the Commission Assistants, and the Directors of the Office of Public Affairs (OPA) and Office of Congressional Affairs (OCA).

In addition to regular meetings with external and internal stakeholders, the staff met on several occasions with staff members of the Senate Committee on Environment and Public Works and the Subcommittee on Clean Air, Climate Change, and Nuclear Safety, at their request, to brief them on the status of safety culture initiative activities.

Finally, the staff briefed the Advisory Committee on Reactor Safeguards (ACRS). The ACRS reviewed the approach and associated IPs and IMCs (except for IP 95003 which was not available at the time of the staff's April 6, 2006 briefing of the ACRS). In a letter issued on April 21, 2006, ACRS concluded that "the staff's proposed approach enhances significantly the ability of the agency to identify and address safety culture issues."

<u>Development of the Approach and Key Features</u>

During the November and December public meetings and in consideration of industry efforts related to safety culture, the staff with the full participation of external stakeholders adopted a definition of safety culture, identified safety culture components and needed ROP enhancements, and developed the overall approach to modify the ROP. The details of the approach were developed over subsequent months. A fuller discussion of these activities is described below.

With regard to the definition of safety culture, the staff adopted the definition that was developed by the International Atomic Energy Agency's (IAEA) International Nuclear Safety Advisory Group (INSAG) as presented in its INSAG-4 publication. INSAG defines safety culture as “that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.” The staff based its decision to use this definition on previous Commission reference to the INSAG definition of safety culture, its adoption by many international regulatory bodies, and its similarity to the Institute of Nuclear Power Operations’ (INPO) definition.

The staff based its identification of safety culture components on a wide variety of sources including documents from INPO and the Electric Power Research Institute (EPRI), academic references, and safety culture documents developed by international regulatory bodies and organizations such as the IAEA and the Nuclear Energy Agency (NEA), as well as staff knowledge and experience. The staff reviewed INPO’s “Principles for a Strong Nuclear Safety Culture” and other safety culture attributes embedded in INPO’s Performance Objectives and Criteria functional areas and compared them with IAEA’s safety culture characteristics and staff’s draft safety culture components. The staff concluded that INPO’s safety culture principles and attributes are comprehensive but noted that INPO’s focus is on excellence, while the NRC’s focus is on ensuring adequate protection of public health and safety. This is reflected in differences in terminology between the NRC safety culture components and some of INPO’s safety culture attributes, which were not written from a regulatory perspective.

As a result of stakeholder feedback, the staff eliminated certain of its components and revised others to provide terminology similar to that used by the industry and thereby support a common understanding of the safety culture components. This resulted in a final set of safety culture components and their descriptions.

Following identification of the safety culture components, the staff and the stakeholders identified those areas of the ROP that warranted enhancements. The staff and stakeholders then identified and discussed ten options proposed by stakeholders and staff for modifying the ROP to address the needed enhancements. This resulted in staff and stakeholder convergence on one of the options (referred to as Option G).

The chosen approach provides a graded regulatory response to plant performance issues so that the regulatory response increases as licensees move to the right in the ROP action matrix, as is currently the case. Key features of the approach to provide a greater focus on safety culture include the following:

- Inspector development of findings and the assessment of performance deficiencies for cross-cutting aspects are consistent with current practice.

- The existing cross-cutting areas were revised to reflect the safety culture components.

- Inspection Manual Chapter 0612, “Power Reactor Inspection Reports,” was revised to reference IMC 0305 Section 06.07.c. to ensure that, when findings with cross-cutting aspects are identified, language is used that parallels the descriptions of the cross-cutting area components in IMC 0305.

- IP 71152, "Identification and Resolution of Problems," was revised to modify the existing guidance for inspectors to assess the effectiveness of the corrective action program, the use of operating experience information and the results of independent and self-assessments. The suggested inspector questions in Appendix 1 were also revised to better assess the licensee's safety conscious work environment.

- The event response procedures IP 71153, "Event Follow-up," IP 93812, "Special Inspection," and IP 93800, "Augmented Inspection Team," were revised to direct inspection teams to also consider contributing causes related to the safety culture components as part of efforts to fully understand the circumstances surrounding an event and its probable causes.

- The range of regulatory actions commensurate with the significance of the Performance Indicator and inspection results was revised as follows:

    - For the third consecutive assessment letter identifying the same substantive cross-cutting issue, the staff modified Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," to provide an option for NRC to request that the licensee perform an assessment of safety culture.

    - For licensees in the Regulatory Response Column, the staff modified IP 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," to verify that the licensee's root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components.

    - For licensees in the Degraded Cornerstone Column, the staff modified IMC 0305, "Operating Reactor Assessment Program," to provide the expectation that the licensee's evaluation of the root and contributing causes determines whether deficient safety culture components caused or significantly contributed to the risk-significant performance issues. The revised IMC 0305 will allow NRC to request the licensee to complete an independent assessment of safety culture if NRC determines that the licensee did not recognize that safety culture components caused or were a significant contributor to the risk-significant performance issues. The staff also modified IP 95002, "Supplemental Inspection Procedure for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," to require inspectors to independently determine whether one or more safety culture component deficiencies caused or contributed significantly to the risk-significant performance issues.

    - For licensees in the Multiple/Repetitive Degraded Cornerstone Column, the staff modified IMC 0305 to provide the expectation that the licensee perform an independent assessment of its safety culture. The staff modified IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstone or Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," to require the staff to (1) assess the licensee's independent evaluation of their safety culture and (2) independently perform an assessment of the licensee's safety culture.

A full description of the approach including the safety culture components and specific enhancements to the IPs and IMCs are described in Enclosure 2. The first description of the approach to enhance the ROP was developed by the Nuclear Energy Institute in December 2005 in order to capture the main features of the approach as it was discussed in the November 2005 public meeting. Enclosure 2 has undergone several revisions by the staff to reflect the changes to the approach as it has evolved.

Current Status

All of the IPs and IMCs described above have been revised based on the approach. The staff provided the revised IPs and IMCs to external stakeholders for comment and has addressed their comments. These IPs and IMCs are scheduled to be approved by the end of May 2006 and issued prior to July 1, 2006 (with the exception of IP 95003 which will be approved and issued prior to July 1, 2006).

With regard to IP 95003, the changes to this procedure are extensive. It provides new guidance on evaluating the licensee's independent assessment of safety culture and on conducting an independent NRC evaluation of the licensee's safety culture. A draft revision to IP 95003 was issued for external stakeholder review and comment, and the staff is making final changes to address input received. This procedure applies to plants in the multiple/repetitive degraded cornerstone column of the action matrix. It is infrequently conducted and only would be implemented after the licensee has completed its root cause and extent of condition investigations and an independent assessment of safety culture. Because of the extensive changes and recognizing the need for finalizing this procedure was less immediate than for the more routinely implemented IPs and IMCs, the staff allowed for a longer period for development and issuance of IP 95003 to change this procedure. However, the IP will be issued prior to the July implementation of the ROP changes.

The staff developed "read and sign" computer-based training (CBT) for inspectors, and all inspectors are expected to have completed the CBT before the Spring 2006 regional inspector counterpart meetings. The staff also has developed training for regional inspectors and managers, which is being provided at the counterpart meetings. This training includes an orientation on safety culture, discussion of changes to the ROP, and illustrative case studies. The training will be completed by the inspectors prior to the July implementation of the safety culture modifications to the ROP.

Remaining Key Activities and Schedule

The staff will implement the changes to the ROP beginning in July 2006 and will assess the changes during the first 18 months of implementation (July 2006 – December 2007). During this period, the staff will also use several means to obtain stakeholder feedback including the monthly ROP working group meetings and the Regional Utility Group meetings.

The industry and the staff are discussing holding industry-sponsored workshops this fall to discuss the changes to the ROP as well as any transition issues.

The staff will develop a RIS to provide background and information to licensees on the enhancements to the ROP and will include staff's plans to address transition issues related to the implementation of the revised IPs and IMCs. In addressing the transition issues, the staff will be consistent with the way the ROP has handled other transition issues, as well as factoring in some of the comments received through the ROP and safety culture public meetings. In the past, the staff has implemented ROP changes going forward, and has not looked back on how those changes would have affected previously completed inspection or assessment outcomes. The staff plans to issue the RIS by the end of July 2006.

The staff is working with the TTC to incorporate aspects of the safety culture initiative into training for inspectors. The staff is also developing just-in-time training for IP 95003 team members. In addition, the staff will provide follow-up training and lessons learned at the Fall 2006 regional counterpart meetings.

The staff will review procedures (e.g., IMC 0350 "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns") to determine if modifications are needed to them based on the enhanced ROP.

Safety Culture and Safety Management

As directed by the Commission, the staff considered "if the cross-cutting issues in the enhanced ROP treatment may be more appropriately labeled Safety Management rather than Safety Culture." The staff investigated the use of the term "safety management" and found that it has been applied in the international community (albeit using different definitions) but not typically in the domestic nuclear industry. The INSAG definition of the safety management system as provided in the INSAG-13 report, "Management of Operational Safety in Nuclear Power Plants," is that "the safety management system comprises those arrangements made by the organization for the management of safety in order to promote a strong safety culture and achieve good safety performance." Safety management focuses on the readily visible characteristics of safety culture and includes such features as the definition of safety requirements and organization; planning, control, and support; implementation; and audit, review, and feedback. Safety culture includes these features of safety management, as well as encompassing features that are not readily visible such as basic assumptions and beliefs of both managers and individuals, which may be at the root cause of repetitive and far-reaching safety performance problems.

Therefore, the staff concluded that the term "safety culture" better captures what needs to be addressed throughout the ROP (i.e., the baseline, assessment, event response, and most especially in the supplemental inspection program of the ROP). A fuller explanation of the basis for this conclusion is provided in Enclosure 3.

RESOURCES:

Approximately 1.6 FTE in the reactor safety inspection sub-program is needed each year to implement this program. No resources are necessary for fiscal year (FY) 2006 due to implementation beginning very late in FY 2006. NRR and RES have resources budgeted for FY 2007 for continued development and program evaluation activities. The FY 2007 resources for OE will be reprogrammed from the allegations program if necessary, potentially delaying or reducing the amount of programmatic oversight and support. For FY 2008, the resources were included in the reactor safety inspection program's FY 2008 request and will be addressed in the planning, budgeting, and performance management (PBPM) process. The staff's estimated inspection resource by office are provided in the table below.

Office	FY 2007	FY 2008
	FTE	FTE
OE	0.2	0.2
RES	0.2	0.2
NRR	1.2*	1.2*
Total	1.6	1.6

* This includes 1.0 regional inspection staff, budgeted by NRR

COMMITMENT:

The staff will update the Commission on the status of this effort, including the results of the implementation assessment, in the Commission Paper on the Reactor Oversight Process Self-Assessment for Calendar Year 2007.

CONCLUSION:

The staff's planned approach is within the framework of the ROP. The staff was guided by the regulatory principles that guided the development of the ROP so that overall assessments of licensee performance remain transparent, understandable, objective, predictable, risk-informed, and performance-based. Further, the safety culture initiative approach satisfies the original objectives:

- To provide better opportunities for the NRC staff to diagnose safety culture weaknesses and take appropriate actions before they result in a degraded cornerstone, which is accomplished through the enhancement of IP 71152 and the treatment of cross-cutting areas.

- To provide the NRC staff with a process to determine the need to specifically evaluate a licensee's safety culture after performance problems have resulted in a licensee being placed in the degraded cornerstone column of the action matrix, which is accomplished through the enhancement of IP 95002 and IMC 0305.

- To provide the NRC staff with a structured process to assess the licensee's safety culture assessment and to independently conduct a safety culture assessment for a licensee in the multiple/repetitive degraded cornerstone column of the action matrix, which is accomplished through the enhancement of IP 95003.

The staff seeks to ensure that the enhancements to the ROP in fact meet the original objectives and that there are no unintended negative consequences. Therefore, an initial implementation period of 18 months is planned, and the staff will assess the changes through the ROP self-assessment process and through opportunities provided to stakeholders for feedback on the changes to the ROP. The staff will make modifications, as needed. The staff will inform the Commission of the results of this initial implementation assessment through the Calendar Year 2007 ROP self-assessment paper.

COORDINATION:
The Office of the General Counsel reviewed this package and has no legal objection. The Chief Financial Officer reviewed this package and determined that it has no financial impact.

/RA/
Luis A. Reyes
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for Operations

Enclosures:
1. Commission Direction in SRM/SECY-04-0111 and SRM/SECY-05-0187
2. Summary of the Reactor Oversight Process' Safety Culture Approach
3. Safety Culture Versus Safety Management

The staff seeks to ensure that the enhancements to the ROP in fact meet the original objectives and that there are no unintended negative consequences. Therefore, an initial implementation period of 18 months is planned, and the staff will assess the changes through the ROP self-assessment process and through opportunities provided to stakeholders for feedback on the changes to the ROP. The staff will make modifications, as needed. The staff will inform the Commission of the results of this initial implementation assessment through the Calendar Year 2007 ROP self-assessment paper.

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COMMISSION DIRECTION IN SRM/SECY-04-0111 AND SRM/SECY-05-0187

SRM/SECY-04-0111: "Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture"

< Enhance the Reactor Oversight Process (ROP) treatment of cross-cutting issues to more fully address safety culture.

< Continue to monitor industry efforts to assess Safety Culture and ensure the Commission remains informed of industry efforts and progress.

< Develop tools that allow inspectors to rely on more objective findings.

< Consider including enhanced problem identification and resolution initiatives as part of this effort.

< Ensure that the inspectors are properly trained in the area of Safety Culture.

< Consider if the cross-cutting issues in the enhanced ROP treatment may be more appropriately labeled Safety Management rather than Safety Culture.

< In making any changes, the staff should follow the established processes for revising the ROP, in particular the process for involving stakeholders.

< Include as part of its enhanced inspection activities for plants in the Degraded Cornerstone Column (referred to as Column 3) of the ROP Action Matrix, a determination of the need for a specific evaluation of the licensee's Safety Culture. The staff should interact with our stakeholders to develop a process for making the determination and conducting the evaluation.

< Continue to monitor developments by foreign regulators.

SRM/SECY-05-0187: "Status of Safety Culture Initiatives and Schedule for Near Term Deliverables"

< Continue to interact with external stakeholders and build from enhancements already made to the ROP in response to the Davis-Besse Lessons Learned Task Force.

< Develop a process for determining if an evaluation of safety culture is warranted when a plant falls into the degraded cornerstone column of the ROP action matrix.

< Keep the Commission offices fully and currently informed of the status of this activity.

< Inform the Commission offices of the key elements of the process before finalizing it.

< Complete requisite training of inspectors on the enhancements to address safety culture.

< Document significant changes to the ROP addressing safety culture in the ROP guidance documents and/or basis documentation.

< Ensure that the resulting modifications to the ROP are consistent with the regulatory principles that guided the development of the ROP.

SUMMARY OF THE REACTOR OVERSIGHT PROCESS' SAFETY CULTURE APPROACH

Introduction

The Commission has long recognized the importance of safety culture as reflected in the development of the inspection program and as it has evolved. The Davis-Besse event reemphasized the importance of safety culture and demonstrated that significant problems can occur as a direct result of safety culture weaknesses that are not recognized and addressed early.

Since the Davis-Besse event occurred, the staff has implemented several improvements to the Reactor Oversight Process (ROP) that relate to safety culture. These improvements include: revisions to the plant assessment process to provide more specific guidance for use in identifying the existence of substantive cross-cutting issues in the areas of human performance and problem identification and resolution; revisions to the baseline (or routine) inspection procedure on "Identification and Resolution of Problems" to require the resident inspector to perform a screening review of each item entered into the corrective action program so as to be alert to conditions such as repetitive equipment failures or human performance issues that might warrant additional follow-up, and to require a semi-annual review to identify trends that might indicate the existence of a more significant safety issue; revision to another inspection procedure to include deferred modifications as one of the areas an inspector can assess; and creation and implementation of a web-based training course for inspectors and managers based on the Columbia Space Shuttle accident which illustrated, for example, the importance of maintaining a questioning attitude toward safety and how issues concerning an organization's safety culture can lead to technological failures.

These changes provide insights into a station's safety culture while appropriately focusing on licensee equipment performance within the scope of the existing baseline inspection program.

In SECY 04-0111, "Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture," dated July 1, 2004, the staff provided options for addressing oversight of a licensee's safety culture including safety conscious work environment. In an August 30, 2004, Staff Requirements Memorandum (SRM) SECY-04-0111, the Commission provided direction to guide the staff's activities to enhance the ROP to more fully address safety culture.

In part, the SRM directed the NRC staff to:

- continue to monitor industry efforts to assess Safety Culture and ensure the Commission remains informed of industry efforts and progress

- enhance the ROP's treatment of cross-cutting issues to more fully address Safety Culture

- ensure that the inspectors are properly trained in the area of Safety Culture

- in making any changes, the staff should follow the established processes for revising the ROP, in particular the process for involving stakeholders

- include as part of its enhanced inspection activities for plants in the Degraded Cornerstone Column (referred to as Column Three) of the ROP Action Matrix, a determination of the need for a specific evaluation of the licensees Safety Culture. The staff should interact with our stakeholders to develop a process for making the determination and conducting the evaluation

A subsequent SRM-SECY-05-0187, "Status of Safety Culture Initiatives and Schedule for Near-term Deliverables," dated December 21, 2005, further directed the staff to:

- continue to interact with external stakeholders;

- build on enhancements already made to the ROP in response to the Davis Besse Lessons Learned Task Force report recommendations;

- identify further improvements to more fully address licensee safety culture;

- keep the Commission offices fully and currently informed of the status of this activity;

- complete requisite training of inspectors on the enhancements to address safety culture by the end of Calendar Year 2006;

- document significant changes to the ROP in the ROP guidance and/or basis documents, and;

- ensure that resulting modifications to the ROP are consistent with the regulatory principles that guided the development of the ROP, such that overall assessments of licensee performance remain transparent, understandable, objective, predictable, risk-informed, and performance-based.

The staff undertook an initiative to respond to the Commission's direction. As part of that initiative, the NRC staff solicited stakeholder input into developing an approach that will enable the agency to detect a declining plant safety culture earlier. This paper outlines the approach that was jointly developed during a public meeting held November 29 - 30, 2005, and was subsequently discussed in public meetings on December 8 and December 15, 2005; January 18, February 2, and February 14, 2006. The approach relies on industry assessments and evaluations by licensees to the extent practical, with NRC staff reviewing results to ensure consistency between these assessments and what has been acknowledged by NRC and stakeholders as those features that are important to safety culture. In addition, the approach allows for the NRC to conduct an independent assessment of a plant's safety culture when there is significant performance degradation. Consistent with the existing ROP framework, the approach supports the regulatory principles that guided the development of the ROP.

Discussion

This paper is divided into two parts, as follows:

- Part I, "Fundamental Items," describes the assumptions upon which this approach is founded, and provides the definition of safety culture and descriptions of safety culture components that have been incorporated into the approach.

- Part II, "Enhanced Reactor Oversight Process Elements," describes how this initiative proposes to enhance the ROP, in terms of baseline inspections, event response inspections, performance assessment, and regulatory responses to degraded performance to more fully address safety culture.

I. Fundamental Items

Assumptions

The approach is based on the following assumptions:

- any issues identified with a licensee's safety culture would be documented in accordance with the current ROP guidelines.

- the titles of the three existing ROP cross-cutting areas (Problem Identification & Resolution, Human Performance, and Safety Conscious Work Environment) will not be changed. However, the contents of each cross-cutting area will be adjusted to better align with the components important to safety culture.

- to the extent possible, the NRC will use existing industry terminology that defines safety culture components.

- the NRC staff will use a graduated or graded response to plant performance issues relative to safety culture, consistent with the existing ROP.

  - the NRC staff will rely on, to the extent practical, licensee and independent assessments of safety culture with NRC review of those assessments.

  - if there is significant performance degradation, the NRC staff will conduct an independent assessment of a licensee's safety culture.

- the approach will remain consistent with the existing ROP framework.

Safety Culture

As part of the staff's interactions with stakeholders, and one of the necessary first steps was to gain agreement on the definition of safety culture. During public meetings in December 2005, there was general agreement that the NRC's proposed use of the International Atomic Energy Agency's (IAEA) International Nuclear Safety Advisory Group (INSAG) definition of safety culture, which had been previously referenced by the Commission, was acceptable and close to the definition that was developed by the Institute of Nuclear Power Operations (INPO).

The INSAG definition was first published in Safety Series No. 75-INSAG-4, "Safety Culture," Vienna, 1991, as "*that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.*"

Participants also agreed that "safety culture" included the following 13 components:

1. Decision-Making
2. Resources
3. Work Control
4. Work Practices
5. Corrective Action Program
6. Operating Experience
7. Self- and Independent Assessments
8. Environment for Raising Nuclear Safety Concerns
9. Preventing, Detecting, and Mitigating Perceptions of Retaliation
10. Accountability
11. Continuous Learning Environment
12. Organizational Change Management
13. Safety Policies

Descriptions of these components are provided in Appendix 1. Safety Culture components 1 - 9 above, termed "cross-cutting components" are aligned with the three cross-cutting areas (i.e., human performance, problem identification and resolution, and safety conscious work environment) and replace the existing cross-cutting subcategories or bins. However, all 13 safety culture components are to be applied in the supplemental inspection program. This distinction was made because:

- the nine cross-cutting components are currently readily accessible through baseline inspection procedures, while the last four safety culture components listed above (i.e., Accountability, Continuous Learning Environment, Organizational Change Management, and Safety Policies) are not.

- each of the nine cross-cutting components is closely aligned with the cross-cutting area with which it is associated, while components 10 - 13 listed above are not closely aligned with a cross-cutting area.

- the cross-cutting components would be considered only when an inspector was considering the cross-cutting aspect of a potential inspection finding or performance deficiency as is done in the existing ROP.

Appendix 2 describes industry activities through which the NRC may gain insights into safety culture at plant sites.

II. Enhanced Reactor Oversight Process Elements

The subsections below describe how this initiative enhances the baseline inspection procedures, performance assessment, cross-cutting areas, substantive cross-cutting issues, event response procedures, and actions for plants in the four columns of the Action Matrix described in Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program: Licensee Response, Regulatory Response, Degraded Cornerstone, and Multiple/Repetitive Degraded Cornerstone," to more fully address safety culture.

Baseline Inspection Procedures

Inspection Procedure 71152, "Problem Identification and Resolution," will continue to:

- provide for early warning of potential performance issues that could result in crossing thresholds in the action matrix,

- help the NRC gauge supplemental response should future action matrix thresholds be crossed,

- allow for follow-up of previously identified compliance issues,

- provide additional information related to cross cutting issues that can be used in the assessment process, and

- determine whether licensees are complying with NRC regulations regarding corrective action programs.

Inspection Procedure 71152, will be enhanced to:

- direct inspectors to take into consideration safety culture components when selecting inspection samples;

- augment the inspection requirements and guidance for evaluating operating experience, the alternative processes for raising concerns, safety conscious work environment, and licensee self-assessments, including periodic assessments of safety culture; and

- modify the existing guidance for inspectors to assess the effectiveness of the corrective action program, the operating experience program, and the licensee's ability to complete self-assessments.

IMC 0612, "Power Reactor Inspection Reports," will be enhanced to be consistent with these changes.

Event Response Procedures

For event response, the NRC staff uses Inspection Procedures 71153, "Event Follow-up;" 93812, "Special Inspection;" and 93800, "Augmented Inspection Team." These procedures will be enhanced to direct inspection teams to be sensitive to causal factors related to safety culture components.

Performance Assessment

As described in IMC 0305, "Operating Reactor Assessment Program," the NRC assesses plant performance continuously and communicates its assessment of plant performance in letters to licensees, typically semi-annually. These assessment letters are available on the NRC website (www.nrc.gov) on the plant performance summary page for each licensee.

Also, as described in IMC 0305, the NRC determines its regulatory response for each licensee in accordance with an Action Matrix that provides for a range of actions commensurate with the significance of the Performance Indicator and inspection results. For a plant that has all of its Performance Indicator and inspection findings characterized as green, the NRC will implement only its baseline inspection program. For plants that do not have all green Performance Indicators and inspection findings, the NRC will perform additional inspections and initiate other actions commensurate with the safety significance of the issues.

Cross-Cutting Areas of Problem Identification & Resolution, Human Performance and Safety Conscious Work Environment

Although the basic structure and titles of the three cross-cutting areas will not change, they will be adjusted to more fully reflect the components that are important to safety culture that can be readily accessed through the baseline inspection program. The table below provides the three cross-cutting areas, the existing subcategories, and the safety culture components which will replace the existing subcategories. These changes will be addressed in IMC 0305, "Operating Reactor Assessment Program." Also, IMC 0612, “Power Reactor Inspection Reports,” will be revised to reference IMC 0305 Section 06.07.c. to ensure that, when findings with cross-cutting aspects are identified, language is used that parallels the descriptions of the cross-cutting area components in IMC 0305.

CROSS-CUTTING AREA	EXISTING SUBCATEGORIES “BINS”	NEW CROSS-CUTTING COMPONENTS
PROBLEM IDENTIFICATION AND RESOLUTION	• Identification • Evaluation • Corrective Action	• Corrective Action Program • Self and Independent Assessments • Operating Experience
HUMAN PERFORMANCE	• Personnel • Resources • Organization	• Decision Making • Resources • Work Control • Work Practices
SAFETY CONSCIOUS WORK ENVIRONMENT	• None	• Environment for Raising Nuclear Safety Concerns • Preventing, Detecting, and Mitigating Perceptions of Retaliation

Substantive Cross-Cutting Issues

As described in IMC 0305, "Operating Reactor Assessment Program," in each assessment meeting (both end-of-cycle and mid-cycle), the NRC will determine whether a substantive cross-cutting issue exists in any cross-cutting area as follows:

- Findings documented in NRC inspection reports are a major input to the assessment process. A documented finding is (1) a more-than-minor[1] NRC-identified or self-revealing issue of concern that is associated with a licensee performance deficiency, and (2) a greater than green licensee-identified finding. A licensee-identified finding of very low (i.e., green) safety significance that are not violations of regulatory requirements are not documented in inspection reports and not used in the assessment process. A finding that is associated with a regulatory requirement is also a violation.

- Each finding is documented in NRC inspection reports in terms of the performance deficiency associated with the finding and the relationship, if any, between the finding and one or more of the cross-cutting areas. A relationship between a finding and a cross-cutting area would exist if a causal factor of the finding is associated with or similar to any part of the description of the components (i.e., a cross-cutting aspect) within that cross-cutting area. (Appendix 1 provides the component definitions that will be used for this purpose by the inspectors). IMC 0612 “Power Reactor Inspection Reports” was revised to ensure that, when findings with cross-cutting aspects are identified, they are aligned with the related safety culture components.

- For the cross-cutting areas of Problem Identification & Resolution and Human Performance, the NRC would identify a substantive cross-cutting issue if all of the following criteria are satisfied:

  < for the current 12-month assessment period, more than three green or safety significant inspection findings have documented cross-cutting aspects in the same cross-cutting area. Observations or violations that are not findings are not considered in this determination.

  < the causal factors for those findings have a common theme.

  < the NRC has a concern with the licensee's scope of efforts or progress in addressing this area's performance deficiency.

- For the Safety Conscious Work Environment cross-cutting area, the NRC would identify a substantive cross-cutting issue if for the current 12-month assessment period,

  < any more-than-minor green or safety significant inspection finding has a documented cross-cutting aspect in the area of Safety Conscious Work

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[1] Inspectors distinguish between minor and more-than-minor findings as described in NRC Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix B, section B-3.

Environment (observations or violations that are not findings are not considered in this determination), or

< the licensee received a chilling-effect letter; or

< the licensee received a letter from the NRC which transmitted an enforcement action with a severity level of I, II, or III, and which involved discrimination; and

< the associated impact on Safety Conscious Work Environment was not isolated, and the NRC has a concern with the licensee's scope of efforts or progress in addressing this area's performance deficiency.

When the NRC informs a licensee that a substantive cross-cutting issue has been identified, the licensee should place that issue into its corrective action program, perform an analysis of causes for the issue, and develop corrective actions. The licensee's completed evaluation may be reviewed by the Region and documented in the next assessment letter.

Substantive cross-cutting issues may be identified by the staff for any licensee, regardless of their position in the Action Matrix. As currently described in IMC 0305, “Operating Reactor Assessment Program,”

> "When the NRC identifies a substantive cross-cutting issue in the mid-cycle or annual assessment letter, the licensee should place this issue into its corrective action program, perform an analysis of causes of the issue, and develop appropriate corrective actions. The licensee's completed evaluation may be reviewed by the regional office and documented in the next mid-cycle or annual assessment letter." (IMC 0305, Section 06.07.e)

For those plants where the same substantive cross-cutting issue has been raised in at least two consecutive assessment letters, the regional office may request that:

- the licensee provide a response at the next annual public meeting;

- the licensee provide a written response to the substantive cross-cutting issues raised in the assessment letters; or

- a separate meeting be held with the licensee.

This provision in IMC 0305 will be enhanced to provide an additional option as follows:

> "Additionally, in the third consecutive assessment letter identifying the same substantive cross-cutting issue, the regional office may also request that the licensee perform an assessment of safety culture. Typically, this evaluation would consist of a licensee self-assessment, unless the recurring substantive cross-cutting issue was associated with deficiencies in the identification or evaluation aspects of the problem identification and resolution program. The regional office should review the safety culture assessment and document the NRC's assessment in the next mid-cycle or annual assessment letter."

Actions in the Licensee Response Column

This initiative proposes no change to actions in the Licensee Response Column.

Actions in the Regulatory Response Column

As currently discussed in IMC 0305, when a licensee's performance falls into the Regulatory Response Column,

> "the licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes."

The licensee enters the corrective actions identified during the above evaluation into the plant's corrective action program.

The licensee's evaluation is reviewed by the NRC during Inspection Procedure 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area." This procedure will continue to provide assurance that:

- the root causes and contributing causes of risk significant performance issues are understood;

- the extent of condition and extent of cause of risk significant performance issues are identified; and

- licensee corrective actions to risk significant performance issues are sufficient to address the root and contributing cause, and to prevent recurrence.

Inspection procedure 95001 will be enhanced to verify that the licensee's root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components.

NRC staff will proceed with all other aspects of the existing process for the Regulatory Response Column as described in IMC 0305.

Actions in the Degraded Cornerstone Column

As currently discussed in IMC 0305, when a licensee's performance falls within the degraded cornerstone column,

- "... the licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes for both the individual and the collective issues."

- "...an independent assessment of the extent of condition will be performed by the region using appropriate inspection procedures chosen from the tables contained in Appendix B to Inspection Manual Chapter 2515."

- the NRC will review the licensee's evaluation using Inspection Procedure 95002, "Supplemental Inspection for One Degraded Cornerstone Or Any Three White Inputs in a Strategic Performance Area."

IMC 0305 will be enhanced to:

- include an expectation for the licensee to ensure that its root-cause evaluation determines whether the plant's performance issues were in any way caused by or contributed to by any component of safety culture, and whether any opportunities exist for improved performance with respect to those components. The licensee should enter into the plant's corrective action program the opportunities for improved performance identified during this assessment. The assessment may be performed by an independent party.

- allow the NRC to request the licensee to complete an independent assessment of safety culture, if the NRC identified and the licensee did not recognize that one or more safety culture components caused or contributed to the risk-significant performance issues.

Inspection Procedure 95002 will continue to:

- provide assurance that the root causes and contributing causes are understood for individual and collective (multiple white inputs) risk significant performance issues.

- independently assess the extent of condition for individual and collective (multiple white inputs) risk significant performance issues.

- provide assurance that licensee corrective actions to risk significant performance issues are sufficient to address the root causes and contributing causes, and to prevent recurrence.

Inspection Procedure 95002 will be enhanced to enable NRC inspectors to independently determine whether any safety culture component caused or contributed significantly to the risk-significant performance issues.

NRC staff would proceed with all other aspects of the existing process for the Degraded Cornerstone Column as described in IMC 0305.

Actions in the Multiple/Repetitive Degraded Cornerstone Column

As currently discussed in IMC 0305, when a licensee's performance falls within the multiple/repetitive degraded cornerstone column,

> "the licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes for both the individual and the collective issues."

This evaluation may consist of a third party assessment.

IMC 0305 will be enhanced to:

- expect the licensee to perform an independent assessment of their safety culture,
- enable NRC inspectors to review that assessment, and
- enable inspectors to independently assess the licensee's safety culture.

In accordance with IMC 0305, the licensee's evaluation will be reviewed by the NRC during Inspection Procedure 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, Or One Red Input." This procedure will continue to:

- provide the NRC additional information to be used in deciding whether the continued operation of the facility is acceptable and whether additional regulatory actions are necessary to arrest declining plant performance.
- provide an independent assessment of the extent of risk significant issues to aid in the determination of whether an unacceptable margin of safety exists.
- independently assess the adequacy of the programs and processes used by the licensee to identify, evaluate, and correct performance issues.
- independently evaluate the adequacy of programs and processes in the affected strategic performance areas.
- provide insight into the overall root and contributing causes of identified performance deficiencies.
- determine if the NRC oversight process provided sufficient warning to significant reductions in safety.

Inspection procedure 95003 will be enhanced to enable NRC inspectors also to:

- independently evaluate the adequacy of the independent assessment of the licensee's safety culture.
- independently assess the licensee's safety culture.

Exercise of the Enhanced Approach for Two Nuclear Power Plants

The staff conducted an exercise to review the inspection and assessment record (2002 Mid-Cycle Assessments - 2004 Mid-Cycle Assessments) for two facilities in an effort to determine how the revisions to the ROP would have handled those plants in order to identify the effectiveness of the changes and to identify whether there may have been any unanticipated consequences. The exercise results indicated that there were no unanticipated consequences and demonstrated the ROP enhancements to be an improvement in terms of earlier agency actions for recurring substantive cross-cutting issues related to safety culture components.

Attachments:
Appendix 1 - Safety Culture Components
Appendix 2 - INPO Lessons-Learned Review

SAFETY CULTURE COMPONENTS

The following safety culture components were developed by the NRC safety culture working group based on the group's research of industry and international documents and the experience of the working group members. The information on safety culture gathered by the working group was screened to ensure that the information in the components is unambiguous, within NRC's regulatory purview, provided insights on the components through existing inspection techniques, and is generally applicable to reactor licensees. The NRC's components were compared to both industry and international safety culture attributes to ensure that the NRC staff fully captured concepts appropriate for NRC oversight. In an effort to utilize language and titles and nomenclature that are common with the industry, the working group compared the NRC's safety culture components to INPO's safety culture attributes and applicable sections of INPO's Performance and Objectives Criteria. Based on this review, some of the NRC's safety culture components were revised to be consistent with INPO's language, where appropriate. To address internal and external stakeholder feedback following the December 8, 2005, December 15, 2005, January 18, 2006, and February 14, 2006 public meetings, the working group further revised the safety culture components to enhance the concepts in the components and utilize language that would better facilitate use of the components under the Reactor Oversight Process (ROP).

The cross-cutting components (i.e., the components of safety culture directly related to one of the cross-cutting areas of Human Performance, Problem Identification and Resolution, and Safety Conscious Work Environment) are described below. The revised inspection procedures and manuals further explain how the NRC staff intend that these components be utilized in the ROP.

HUMAN PERFORMANCE

<u>Decision-Making</u> -Licensee decisions demonstrate that nuclear safety is an overriding priority. Specifically (as applicable):

- The licensee makes safety-significant or risk-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. This includes formally defining the authority and roles for decisions affecting nuclear safety, communicating these roles to applicable personnel, implementing these roles and authorities as designed, and obtaining interdisciplinary input and reviews on safety-significant or risk-significant decisions.

- The licensee uses conservative assumptions in decision making and adopts a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. The licensee conducts effectiveness reviews of safety-significant decisions to verify the validity of the underlying assumptions, identify possible unintended consequences, and determine how to improve future decisions.

The licensee communicates decisions and the basis for decisions to personnel who have a need to know the information in order to perform work safely, in a timely manner.

Resources - The licensee ensures that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically those necessary for:

- Maintaining long term plant safety by maintenance of design margins, minimization of long-standing equipment issues, minimizing preventive maintenance deferrals, and ensuring maintenance and engineering backlogs are low enough to support safety

- Training of personnel and sufficient qualified personnel to maintain work hours within working hours guidelines

- Complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components

- Adequate and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities, and equipment

Work Control -The licensee plans and coordinates work activities, consistent with nuclear safety. Specifically (as applicable):

- The licensee appropriately plans work activities by incorporating:
  - risk insights
  - job site conditions, including environmental, which may impact human performance; plant structures, systems, and components; human-system interface; and radiological safety
  - the need for planned contingencies, compensatory actions, and abort criteria

- The licensee appropriately coordinates work activities by incorporating actions to address:
  - the impact of changes to the work scope or activity on the plant and human performance
  - the impact of the work on different job activities
  - the need for work groups to maintain interfaces with offsite organizations, and communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance
  - the need to keep personnel apprised of work status, the operational impact of work activities, and plant conditions that may affect work activities

- The licensee plans work activities to support long-term equipment reliability by limiting temporary modifications, operator work-arounds, safety systems unavailability, and reliance on manual actions. Maintenance scheduling is more preventive than reactive.

Work Practices - Personnel work practices support human performance. Specifically (as applicable):

- The licensee communicates human error prevention techniques, such as holding pre-job briefings, self and peer checking, and proper documentation of activities. These techniques are used commensurate with the risk of the assigned task, such that work activities are performed safely. Personnel are fit for duty. In addition, personnel do not proceed in the face of uncertainty or unexpected circumstances.

- The licensee defines and effectively communicates expectations regarding procedural compliance.

- The licensee ensures supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported.

PROBLEM IDENTIFICATION AND RESOLUTION

Corrective Action Program (CAP) - The licensee ensures that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance. Specifically (as applicable):

- The licensee implements a corrective action program with a low threshold for identifying issues. The licensee identifies such issues completely, accurately, and in a timely manner commensurate with their safety significance.

- The licensee periodically trends and assesses information from the CAP and other assessments in the aggregate to identify programmatic and common cause problems. The licensee communicates the results of the trending to applicable personnel.

- The licensee thoroughly evaluates problems such that the resolutions address causes and extent of conditions, as necessary. This includes properly classifying, prioritizing, and evaluating for operability and reportability conditions adverse to quality. This also includes, for significant problems, conducting effectiveness reviews of corrective actions to ensure that the problems are resolved.

- The licensee takes actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity.

- If an alternative process (i.e., a process for raising concerns that is an alternate to the licensee’s corrective action program or line management) for raising safety concerns exists, then it results in appropriate and timely resolutions of identified problems.

Operating Experience - The licensee uses operating experience (OE) information, including vendor recommendations and internally generated lessons learned, to support plant safety. Specifically (as applicable):

- The licensee systematically collects, evaluates, and communicates to affected internal stakeholders in a timely manner relevant internal and external OE.

- The licensee implements and institutionalizes OE through changes to station processes, procedures, equipment, and training programs.

Self- and Independent Assessments - The licensee conducts self- and independent assessments of the organization's activities and practices, as appropriate, to assess performance and identify areas for improvement. Specifically (as applicable):

- The licensee conducts self-assessments at an appropriate frequency; such assessments are of sufficient depth, are comprehensive, are appropriately objective, and are self-critical. The licensee periodically assesses the effectiveness of oversight groups and programs such as CAP, and policies.

- The licensee tracks and trends safety indicators which provide an accurate representation of performance.

- The licensee coordinates and communicates results from assessments to affected personnel, and takes corrective actions to address issues commensurate with their significance.

SAFETY CONSCIOUS WORK ENVIRONMENT

Environment For Raising Concerns - An environment exists in which employees feel free to raise concerns both to their management and/or the NRC without fear of retaliation and employees are encouraged to raise such concerns. Specifically (as applicable):

- Behaviors and interactions encourage free flow of information related to raising nuclear safety issues, differing professional opinions, and identifying issues in the CAP and through self assessments. Such behaviors include supervisors responding to employee safety concerns in an open, honest, and non-defensive manner and providing complete, accurate, and forthright information to oversight, audit, and regulatory organizations. Past behaviors, actions, or interactions that may reasonably discourage the raising of such issues are actively mitigated. As a result, personnel freely and openly communicate in a clear manner conditions or behaviors, such as fitness-for-duty issues, that may impact safety and personnel raise nuclear safety issues without fear of retaliation.

- If alternative processes (i.e., a process for raising concerns or resolving differing professional opinions that are alternates to the licensee's corrective action program or line management) for raising safety concerns or resolving differing professional opinions exists, then they are communicated, accessible, have an option to raise issues in confidence, and are independent from management who would in the normal course of activities be responsible for addressing the issue.

Preventing, Detecting, and Mitigating Perceptions of Retaliation - A policy for prohibiting harassment and retaliation for raising nuclear safety concerns exists and is consistently enforced in that:

- All personnel are effectively trained that harassment and retaliation for raising safety concerns is a violation of law and policy and will not be tolerated.

- Claims of discrimination are investigated consistent with the content of the regulations regarding employee protection and any necessary corrective actions are taken in a timely manner, including actions to mitigate any potential chilling effect on others due to the personnel action under investigation.

- The potential chilling effects of disciplinary actions and other potentially adverse personnel actions (e.g., reductions, outsourcing, and reorganizations) are considered and compensatory actions are taken when appropriate.

OTHER SAFETY CULTURE COMPONENTS

The following are other safety culture components which are not associated with the cross-cutting areas. These components when combined with the cross-cutting area components, comprise the safety culture components. Components in this section are considered only during the conduct of the supplemental inspection program, while the cross-cutting area components are considered during the conduct of both the baseline and supplemental inspection programs.

<u>Accountability</u> - Management defines the line of authority and responsibility for nuclear safety. Specifically (as applicable):

- Accountability is maintained for important safety decisions in that the system of rewards and sanctions is aligned with nuclear safety policies and reinforces behaviors and outcomes which reflect safety as an overriding priority.

- Management reinforces safety standards and displays behaviors that reflect safety as an overriding priority.

- The workforce demonstrates a proper safety focus and reinforce safety principles among their peers.

<u>Continuous learning environment</u> - The licensee ensures that a learning environment exists. Specifically (as applicable):

- The licensee provides adequate training and knowledge transfer to all personnel on site to ensure technical competency.

- Personnel continuously strive to improve their knowledge, skills, and safety performance through activities such as benchmarking, being receptive to feedback, and setting performance goals. The licensee effectively communicates information learned from internal and external sources about industry and plant issues.

<u>Organizational change management</u> - Management uses a systematic process for planning, coordinating, and evaluating the safety impacts of decisions related to major changes in organizational structures and functions, leadership, policies, programs, procedures, and resources. Management effectively communicates such changes to affected personnel.

Safety policies - Safety policies and related training establish and reinforce that nuclear safety is an overriding priority in that:

- These policies require and reinforce that individuals have the right and responsibility to raise nuclear safety issues through available means, including avenues outside their organizational chain of command and to external agencies, and participate in the resolution of such issues.

- Personnel are effectively trained on these policies.

- Organizational decisions and actions at all levels of the organization are consistent with the policies. Production, cost and schedule goals are developed, communicated, and implemented in a manner that reinforces the importance of nuclear safety.

- Senior managers and corporate personnel periodically communicate and reinforce nuclear safety such that personnel understand that safety is of the highest priority.

INDUSTRY ACTIVITIES PROVIDING INSIGHT INTO SAFETY CULTURE

The Institute of Nuclear Power Operations (INPO) conducted a lessons-learned review as a result of the Davis-Besse head degradation issue. Sixteen improvement items were identified, covering each of the four cornerstone areas that INPO provides for the nuclear industry (evaluation, training and accreditation, operating experience, and assistance). INPO also issued Significant Operating Experience Report (SOER) 02-4 in 2002 as a result of the Davis-Besse head degradation incident. Each station, per the SOER recommendations, performed an assessment of its safety culture. INPO, through its evaluation process, has evaluated implementation of that recommendation at each licensee station. The SOER further recommended that, going forward, each licensee periodically conduct a safety culture assessment. Although the frequency of these evaluations may vary, these evaluations provide insights into the health of a station's safety culture at each licensee's facility.

INPO developed "Principles for Effective Self Assessment and Corrective Action Programs." This document is an industry standard for conduct of these important programs. Included in the principles for effective self-assessment programs is the following expectation:
Station management verifies that the issues are promptly entered into the corrective action program or other tracking system for resolution. The principles document further states that:
… tracking systems are periodically screened to preclude important problems that should be in the corrective action program from being reported instead to lower-tier tracking systems in which they may receive a lower level of analysis and corrective action. Therefore, issues such as those likely to significantly affect or be driven by a licensee's safety culture would be handled within the licensee's corrective action program. These licensee assessments, as well as the results, are therefore available to the NRC staff during their Problem Identification & Resolution (PI&R) inspections.

In addition to licensee assessments, INPO performs plant evaluations on approximately a two year frequency. These evaluations are a comprehensive, INPO and industry peer team evaluation of plant performance that includes an assessment of the plant's adherence to key safety culture principles and attributes as defined in INPO’s “Principles of a Strong Nuclear Safety Culture” document. This evaluation is performed as part of an assessment of each station's Organizational Effectiveness cross functional area, in accordance with INPO's Performance Objectives and Criteria document.

INPO documents a summary of its evaluation regarding a station's safety culture in the Organizational Effectiveness Area Performance Summary for each plant. INPO's evaluation reports are not public documents. However, per the existing NRC/INPO Memorandum of Understanding, the NRC is afforded the opportunity to review these reports. This review also provides the NRC staff with insights into a plant's safety culture.

SAFETY CULTURE VERSUS SAFETY MANAGEMENT

BACKGROUND

The Commission provided direction to the staff which stated, in part, to enhance the treatment of cross-cutting areas of the Reactor Oversight Process (ROP) to more fully address safety culture. The Commission provided this direction in Staff Requirements Memorandum (SRM) SRM/SECY-04-0111 and more recently in SRM/SECY-05-0187. The major direction in SRM SECY-04-0111 was to:

- “…enhance the Reactor Oversight Process ... treatment of cross-cutting issues to more fully address Safety Culture.”

- “…include as part of its enhanced inspection activities for plants in the Degraded Cornerstone Column (referred to as Column Three) of the [ROP] Action Matrix, a determination of the need for a specific evaluation of the licensee’s Safety Culture…The staff's methodology for using the treatment of cross-cutting issues to more fully address Safety Culture, should require a specific determination for plants in the Degraded Cornerstone Column.”

In addition, the Commission directed the staff to:

- “…consider if the cross-cutting issues in the enhanced [Reactor Oversight Process] treatment may be more appropriately labeled Safety Management rather than Safety Culture.”

This paper addresses the issue of using the term “Safety Management” versus “Safety Culture.”

DISCUSSION

The goal of the U.S. Nuclear Regulatory Commission’s (NRC) safety culture initiative is to enhance the NRC’s Reactor Oversight Process (ROP) to more fully address safety culture. Safety culture is both an *outcome* of safety management and a *determinant* of aspects of safety management.

Safety Culture Definition

“Safety culture” as a term was first coined by the International Atomic Energy Agency (IAEA) International Nuclear Safety Advisory Group (INSAG) in their INSAG-3 Report (1989) following the Chernobyl accident, and has been in wide use since then. INSAG-3 resulted in broad international interest in the expansion and formalization of the concept of "safety culture" in a manner that would be useful for nuclear power plant operators and regulators. In response to this interest, INSAG-4 presented a formal consensus definition developed by the INSAG members.

The staff has adopted the INSAG-4 definition of safety culture as presented in the INSAG-4 Report (1991) which has been historically referenced by the Commission. The INSAG-4 definition is *“that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention* *warranted by their significance."* INSAG-4 also states "*safety culture is both attitudinal as well as structural and relates to both organizations and individuals*."

Safety Management Definition

Safety management is *how* an organization promotes safety culture and achieves safety performance. "Safety Management" as a term was first used formally in 1999 in INSAG-13, which was geared toward nuclear power plant managing organizations as noted in the foreword while INSAG-4 on safety culture was intended for a wider audience including regulators. INSAG-13 states that safety management is "*the structural aspect of safety culture comprises the organization's arrangements for safety, which is commonly described as the safety management system of the organization*… '*Management' is used to mean the administration of the organization*." …"The *safety management system* comprises those *arrangements made by the organization for the management of safety in order to promote a strong safety culture and achieve good safety performance*."

Figure 1 from the INSAG-4 document illustrates the concepts and elements of safety culture as it is commonly defined. Safety management encompasses a subset of the safety culture elements in the figure (e.g., programmatic elements shown to the left of the "management commitment" element, such as the definition of responsibilities and rewards and sanctions) and also reaches beyond safety culture to other organizational aspects of safety performance (e.g., quality assurance programs) which may be outside the scope of the safety culture initiative.

Use of Terms

The NRC has held eight public meetings (August 2005 through February 2006) as part of the agency's safety culture initiative. Meeting participants included representatives from utility companies, nuclear power plants, industry groups, and public interest groups. In the course of the meetings, the participants agreed that there is a common understanding of the term *safety culture*. This term is used by industry organizations in the U.S., such as the Institute of Nuclear Power Operations (INPO) and by international groups, such as IAEA, and the industry and regulatory bodies in other countries, such as Canada, Finland, Germany, Hungary, Spain, Switzerland, and the U.K. Most either use the INSAG-4 definition (e.g., the International Committee on Nuclear Technology in Germany, the transportation industry in the U.S.; the Hungarian regulator, or build on the elements of the INSAG-4 definition (e.g., in Finland.) While the INPO definition is slightly different, it is similar and encompasses the same concepts and elements as the INSAG-4 definition.

Since safety management is a newer term, the practical definitions of safety management are still evolving and there is not as much convergence across groups and nations on the essential elements that comprise safety management. Some regulatory bodies in other countries, e.g., Switzerland and Finland, have begun to use the term safety management. Those who use the term have either adopted the INSAG-13 definition, or built on the INSAG-13 definition.

Based on their review of the various uses of the terms safety culture and safety management, the staff determined that the term *safety management* is not as widely used nor as commonly understood as the term *safety culture* across different groups in the U.S. and internationally.

Most importantly, *safety culture* is the overriding concern and desired outcome of those groups who use the term *safety management*, i.e., promoting and ensuring a strong safety culture is the driving force behind oversight of safety management.

A Model of Safety Culture and Safety Management

Dr. Edgar H. Schein, Sloan Fellows Professor of Management Emeritus, Massachusetts Institute of Technology (MIT), developed a model of culture which IAEA and others adopted for their safety culture model. Figure 2 is representative of Dr. Schein's model which has three levels of organizational culture:

- Level 1: Artefacts (visible): Tangible products, behavior, organization structure and production processes

- Level 2: Espoused values (not visible, but can be elicited): Values and norms, strategies, objectives, philosophy

- Level 3: Basic assumptions: Unconscious and self-evident beliefs and assumptions

The concept of safety culture encompasses all three levels of Schein's model, including basic assumptions. Safety management elements primarily fall into Dr. Schein's Level 1 – the regulations and procedures (the *how* safety is achieved), Level 2 - policy statements, and the outcomes of safety management that can be observed in behaviors. Although an organization's basic assumptions are less amenable to direct regulatory oversight than visible artefacts and espoused values, problems at Level 3 can serve as the "root causes" that may lead to repetitive and far-reaching safety performance problems. The regulator can gain insights into the third level for plants exhibiting declining performance, through more intrusive oversight in supplemental reactive inspection programs. For example, through evaluating the licensee's safety culture assessment and through NRC's independent assessment of the licensee's safety culture in the conduct of IP 95003.

The NRC's Proposed Components of Safety Culture

The NRC's proposed components of safety culture contain a mix of Level 1: performance outcomes; Level 2: policies, programs and processes, i.e., "Corrective Action Program" and formal "Safety Policies" (management); and Level 3: attitudes (culture and outcomes). *Safety culture* encompasses all 13 safety culture components developed in the initiative for the enhanced ROP. *Safety management* encompasses only a subset of the safety culture components. For example, the safety-conscious working environment components are aspects of safety culture but not safety management.

CONCLUSION

Based on consideration of the factors described above, the staff believes that the term safety culture versus safety management should be used in the NRC's enhanced ROP for the following reasons:

(1) The term, “safety culture,” is used internationally with general convergence on the elements of the definition, as first established in the INSAG-4 Report. The term, “safety management,” is used in a subset of countries, but is applied differently and there is not yet convergence on the elements of safety management. Importantly, the countries and organizations that use the term “safety management” are ultimately concerned about “safety culture” as the outcome of safety management.

(2) Safety culture encompasses the three levels of Dr. Schein’s model of organizational culture (the basis for most safety culture definitions), whereas “safety management” encompasses only the top level (artefacts) and part of the middle level (espoused values). The lowest level, “basic assumptions,” can be of greatest concern to the NRC and other regulatory and industry bodies, because often this level encompasses the root causes that may lead to repetitive and far-reaching safety problems. However, this level is not addressed by the most common definitions of “safety management” (cf. INSAG-13).

(3) Currently the term “safety culture” encompasses all of the NRC’s components of safety culture, while only a subset of the components fit under the term “safety management.” This is an important distinction since all the safety culture components would need to be evaluated in the supplemental inspection program.

(4) Some stakeholders have expressed concern about introducing new terms and concepts. Safety culture is a term in common use in the US nuclear power industry and has achieved a common understanding through INPO’s Principles for a Strong Nuclear Safety Culture and other industry methodologies.

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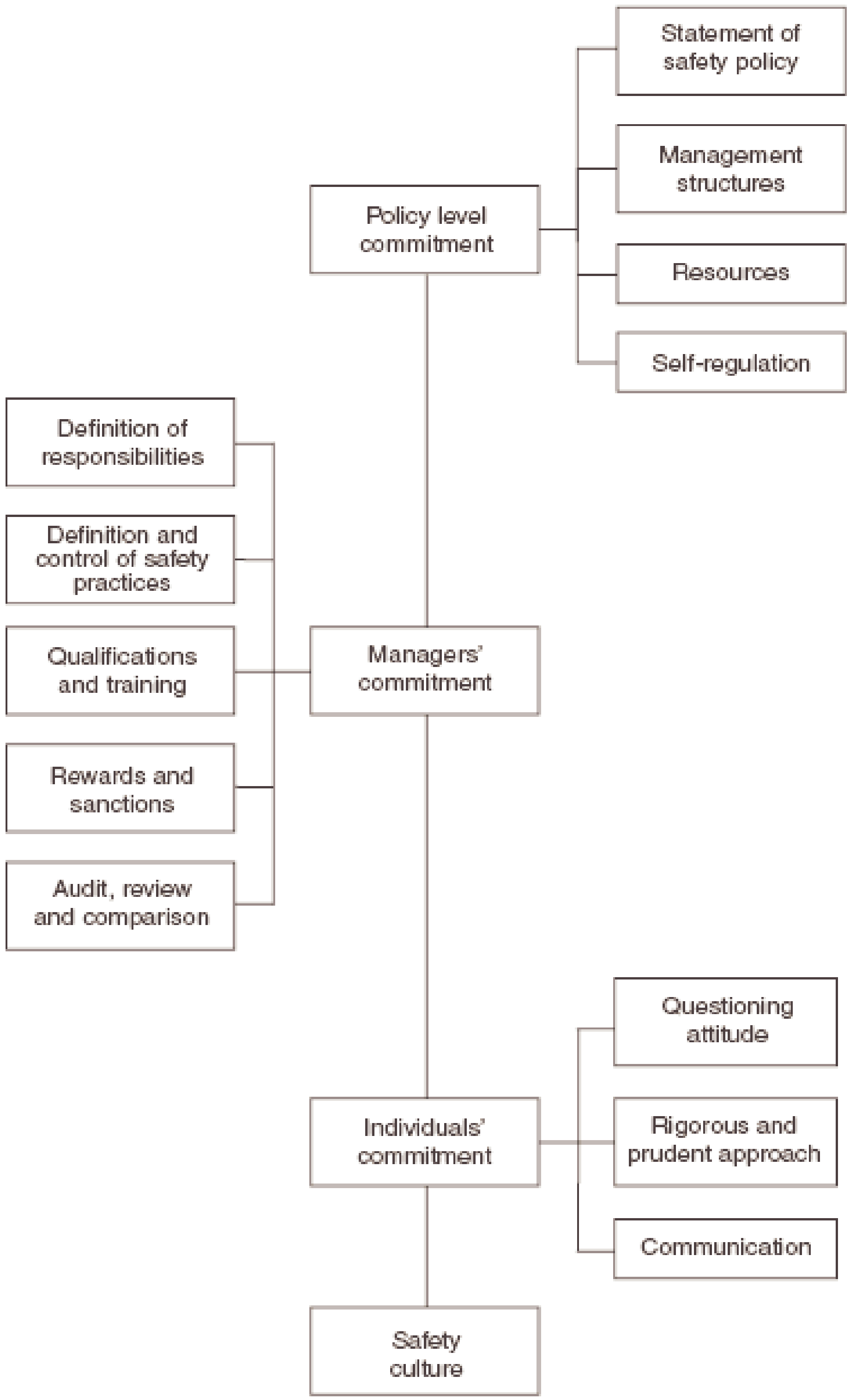

Figure 1. Illustration of the Presentation of Safety Culture (INSAG-4, 1991)

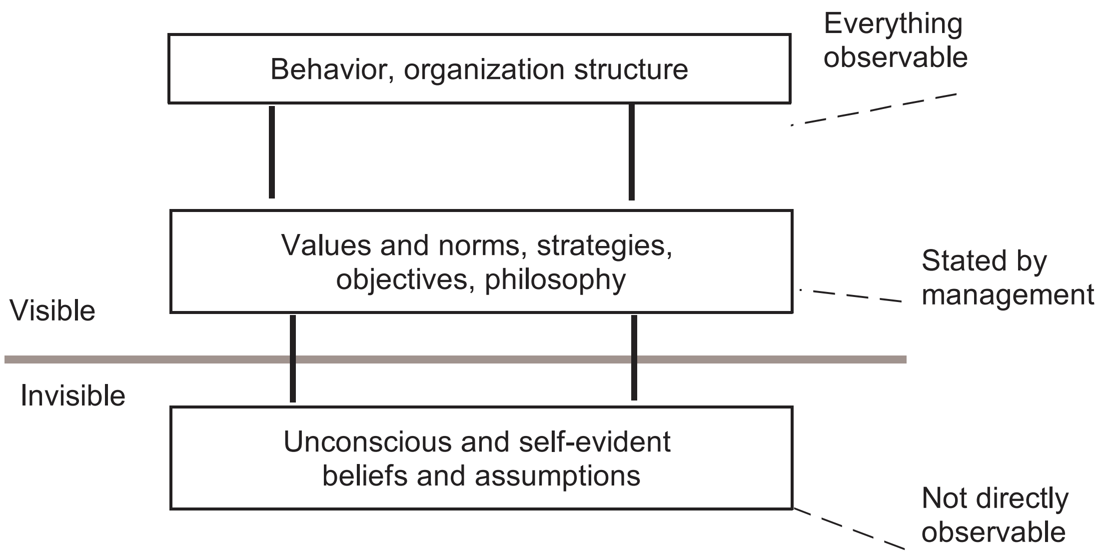

Figure 2. The levels of organization culture