NRC INSPECTION MANUAL

IPAB

MANUAL CHAPTER 0308

REACTOR OVERSIGHT PROCESS (ROP)
BASIS DOCUMENT

0308-01 PURPOSE

To describe the basis for the significant decisions reached by the U.S. Nuclear Regulatory Commission (NRC) staff during the development and implementation of the Reactor Oversight Process (ROP) for operating commercial nuclear power plants. This document shall serve as the source information for all applicable program documents such as manual chapters, performance indicator guidance, and assessment guidance.

0308-02 OBJECTIVES

02.01 To discuss significant developmental steps and decisions reached.

02.02 To describe in general how the processes work and why they are setup the way they are.

02.03 To summarize the history of, and reasons for, significant changes made to the oversight processes.

02.04 To explain those significant attributes that were considered but not used in the ROP, and the basis for the decision not to include them in the process.

0308-03 DEFINITIONS

None stated.

0308-04 RESPONSIBILITIES AND AUTHORITIES

None stated.

0308-05 GENERAL REQUIREMENTS

05.01 Introduction.

On April 2, 2000, the NRC implemented a new ROP at all operating commercial nuclear power plants. The objectives of the staff in developing the various components of this new oversight process were to provide tools for inspecting and assessing licensee performance in a manner that was more risk-informed, objective, predictable, and understandable than the previous oversight processes. The ROP was also developed to meet the four agency performance goals to: (1) maintain safety, (2) increase openness, (3) make NRC activities and decisions more effective, efficient, and realistic, and (4) reduce unnecessary regulatory burden.

In developing the new ROP, many aspects of the old oversight process, such as the inspection program, assessment process, and enforcement policy were revised to meet the above stated objectives and be better integrated and streamlined. Additionally, several new oversight processes were developed, such as performance indicators (PIs) and a significance determination process (SDP) for inspection findings. An overview of the ROP and how each of the individual processes interact can be seen in Exhibit 1. The following discussion provides the background for how the ROP was developed, the basis for many of the key attributes of the new oversight process, and the basis for many aspects of regulatory oversight that were considered, but not included in the ROP.

Additional detail regarding the development and basis for each of the individual oversight processes is included in the appendices to this document. Attachment 1 discusses the PIs and describes the basis for selecting the initial set of PIs and their thresholds, and how the PIs were benchmarked. IMC 308 Attachment 2 describes the Inspection Program and discusses the concepts of the baseline and supplemental inspections. IMC 308 Attachment 3 discusses the basis for the different SDPs that have been developed to evaluate the safety significance of inspection findings. IMC 308 Attachment 4 discusses how the Assessment Program was developed to identify the appropriate NRC actions to take based on the PIs and inspection findings generated. IMC 308 Attachment 5 describes the significant changes made to the Enforcement Policy to support the ROP.

05.02 <u>Background</u>.

In a Staff Requirements Memorandum (SRM) dated June 28, 1996 (Ref. 1), the Commission directed the staff to assess the Senior Management Meeting (SMM) process and evaluate the development of indicators that can provide a basis for judging whether a plant should be placed on or deleted from the NRC Watch List. In response to the Commission's request, a study of the effectiveness of the SMM process was completed on December 30, 1996, by the Arthur Anderson Company (Ref. 2). On April 2, 1997, the staff issued SECY-97-072 (Ref. 3) to inform the Commission of the staff's plans to address the recommendations made by the Arthur Andersen Company. On June 24, 1997, the Commission issued SRM M970424B (Ref. 4) in which it approved the staff's plan to develop improvements to the SMM process.

In parallel with the efforts of the now former Office for Analysis and Evaluation of Operational Data (AEOD) to evaluate improvements to the SMM process, several SRMs directed the staff to improve the objectivity, accuracy, and efficiency of the current assessment process and to evaluate the efficacy of defining and formalizing a unified licensee performance assessment program that integrates the various separate processes being utilized. On June 6, 1997, the staff issued SECY-97-122 (Ref. 5) to inform the Commission of the staff's plans to perform an integrated review of the assessment processes (IRAP), including plant performance reviews (PPRs), systematic assessments of licensee performance (SALPs), and SMMs. On August 19, 1997, the Commission issued SRM 9700238 (Ref. 6) which approved the staff's plans to perform the integrated review.

An IRAP team was assembled with representatives from each regional office, AEOD, the Office of Enforcement (OE), the Office of the Executive Director for Operations, the Office of Nuclear Reactor Regulation (NRR) Inspection Program Branch, and the NRR Division of Reactor Projects (now the Division of Licensing Project Management). The team members included a cross section of experience represented by Deputy Division Directors, Branch Chiefs, Project Managers, and staff with recent regional inspection experience. The IRAP team took a process re-engineering approach to identify those objectives, attributes, and activities that a new assessment process would need to adequately assess licensee performance and to identify the sources of information necessary to support the assessment. The team evaluated the current assessment processes, such as the SALP, PPR, and the SMM, using continuous quality improvement techniques to determine which attributes may be retained to support the new process. The inspection and enforcement programs were assumed to be implemented "as-is" for the integrated review, while any necessary changes to these programs resulting from this effort will be evaluated separately following the integrated review.

On March 9, 1998, the staff issued SECY-98-045 (Ref. 7) which forwarded the staff's recommendation for a new integrated assessment process. The fundamental concepts that formed the basis of the IRAP proposal were: (1) inspection findings provided the basis for the assessment, (2) inspection findings would be categorized by performance template areas and would be scored according to safety significance, (3) assessment would be accomplished by totaling the scores in each template area and comparing these scores against threshold values, and (4) NRC actions would be taken based on a decision model.

On June 30, 1998, the Commission issued the SRM for SECY-98-045 (Ref. 8), in which the Commission expressed concerns with: (1) the apparent use of enforcement as a "driving force" for the assessment process, (2) the quantitative scoring of plant issues matrix (PIM) entries, and (3) the use of color coding to define performance rating categories. However, the Commission did approve the solicitation of public comment on the IRAP proposal, and requested the staff to: (1) provide a recommendation for changes to the assessment process, (2) address regional consistency and equitable treatment of plants receiving varying levels of inspection effort, and (3) include conceptual changes to the inspection program needed to conform with the new assessment process.

In parallel with the staff's development of the IRAP proposal, the industry developed an independent proposal for improving the oversight process, documented in a draft white paper (Ref. 9). This effort, led and coordinated by the Nuclear Energy Institute (NEI), resulted in a concept that was fundamentally and philosophically different from the IRAP proposal. This approach established tiers of licensee performance based on maintaining the barriers to radionuclide release, minimizing events that could challenge the barriers, and ensuring that systems can perform their intended functions. Performance in these tiers would be measured through reliance on high-level, objective indicators with thresholds set for each indicator to form a utility response band, a regulator response band, and a band of unacceptable performance.

In response to the NEI proposal, Commission comment on the IRAP proposal, and comments made at the July 17, 1998, Commission meeting with public and industry stakeholders, the staff set out to develop a single set of recommendations for making improvements to the regulatory oversight processes.

The IRAP public comment period and a series of public meetings were used to facilitate internal and external stakeholder input into the development of these recommendations.

The 60-day IRAP public comment period, which ended on October 6, 1998, was used to seek comment on improvements to the assessment process. As part of the public comment period, the staff sponsored a 4-day public workshop from September 28 through October 1, 1998, to interact with the industry and public to obtain and evaluate input on improving the regulatory oversight processes. During the workshop a consensus was reached on the overall philosophy for regulatory oversight and general agreement was achieved among workshop participants on the defining principles for the oversight processes.

After the workshop, the staff began several short-term activities to continue developing the improvements to the regulatory oversight process that had been initiated at the workshop. All of these activities were coordinated and integrated and involved broad participation from all four regions, NRR, OE, the Office of Nuclear Regulatory Research (RES), and AEOD. The staff selected to participate in these activities were agency experts in various aspects of regulatory oversight, such as risk analysis, use of PIs, inspection, and assessment techniques. Each of these activities also involved frequent interaction with the industry and the public during the development of recommended improvements.

Three task groups were formed to develop these recommendations: a technical framework task group, an inspection task group, and an assessment task group. The technical framework task group was responsible for completing the regulatory oversight structure and for identifying the PIs and appropriate thresholds that could be used to measure performance. The inspection task group was responsible for developing the scope, depth, and frequency of a risk-informed baseline inspection program that would be used to supplement and verify the PIs. The assessment process task group developed methods for integrating PI and inspection data, determining NRC action based on assessment results, and communicating results to licensees and the public. OE activities to improve the enforcement process were coordinated with these three task groups to ensure that enforcement process changes were properly evaluated in the framework structure, and that changes to the inspection and assessment programs were integrated with changes to the enforcement program.

On January 8, 1999, the staff issued SECY-99-007 (Ref. 10) forwarding the staff's recommendations for a revised ROP for commercial nuclear power plants. These recommendations consisted of a framework for regulatory oversight that established seven cornerstones of safety. Fundamental to this concept was that licensee performance that met the objectives and key attributes of each of these cornerstones would provide reasonable assurance that public health and safety was maintained.

In the ROP, licensee performance within each cornerstone is measured by a combination of PIs and inspection results. PIs were developed for each of the cornerstones to provide an objective indication of licensee performance. A risk-informed baseline inspection program was developed to both independently verify the PIs and to inspect those aspects of licensee performance not adequately covered by a PI. The risk-informed baseline inspection program established the minimum inspection effort that all licensees would receive, regardless of their performance.

Risk-informed thresholds were developed for both the PIs and inspection findings to establish performance bands. These performance bands provide for increased regulatory action as licensee performance degrades, as indicated by crossing more risk significant thresholds. A key aspect of using performance thresholds is that it establishes a level of licensee performance that does not warrant additional NRC involvement beyond the baseline inspection program.

The assessment process was redesigned to be more streamlined and objective by using the PIs and inspection findings as assessment inputs and applying an Action Matrix, Figure 1 of IMC 308 Attachment 4, to determine the appropriate follow-up to indications of degrading licensee performance. The enforcement process was also revised to be better integrated and consistent with the inspection program and assessment process.

On March 22, 1999, the staff issued SECY-99-007A (Ref. 11) which forwarded to the Commission additional information on the concepts for the ROP, and presented the staff's plans for a 6-month pilot of the revised oversight processes at two sites per region. On June 18, 1999, the SRM on SECY-99-007 and SECY-99-007A (Ref. 12) was issued which approved the scope and concepts for the ROP and approved the staff plan for the pilot program.

The 6-month pilot program for the ROP was conducted at two sites per region from May 30, 1999, to November 27, 1999. The pilot program was conducted in accordance with the guidelines and procedures forwarded by memorandum from the Director, NRR to the four Regional Administrators (RAs), dated May 20, 1999 (Ref. 13). The sites participating in the pilot program were:

Region I	Region II	Region III	Region IV
Salem/Hope Creek	Shearon Harris	Prairie Island	Fort Calhoun
FitzPatrick	Sequoyah	Quad Cities	Cooper

The purpose of the pilot program was to apply the ROP and identify lessons learned so that the various processes and procedures could be refined and revised as necessary prior to initial implementation. The objectives of the pilot program were: (1) to exercise the various components of the ROP to evaluate whether or not they could function efficiently, (2) to identify significant process and procedure problems and make appropriate changes prior to initial implementation, and (3) to the extent possible, evaluate the effectiveness of the new process. Pilot program criteria were established to evaluate the results of implementing the ROP at the pilot plants.

In addition to evaluating the new process against these pilot program criteria, the staff employed a number of methods to obtain internal and external stakeholder feedback and comments during the pilot program.

Internal feedback and comments from NRC staff were obtained using various methods. Weekly teleconferences were held with regional management and biweekly teleconferences with the pilot program resident inspectors to solicit feedback. Monthly counterpart meetings were held with the regional Division Directors and Executive Forum meetings were periodically conducted with the four Deputy RAs to solicit feedback and comments on the ROP. Inspection procedure and oversight process feedback forms were developed and used during the pilot program for regional staff to document questions and concerns on the various components of the ROP. Comments from these feedback forms were utilized by the staff in making needed modifications to procedures as the pilot program progressed. Finally, an internal stakeholder survey of the RAs and staff who participated in the pilot program was conducted at the end of the pilot to gather additional insights to be considered while evaluating the pilot program lessons learned.

Public comment was solicited on the ROP and the results of the pilot program by a Federal Register notice (FRN) (Ref. 14). The FRN established a public comment period that ended on December 31, 1999, and included a questionnaire to focus public comment on specific topics. This questionnaire requested comment and feedback on the ROP's ability to meet the four agency outcome measures, and also requested feedback and comments on topics such as the role of positive inspection findings in the ROP and the need to develop overall assessment ratings for nuclear power plants.

To keep local public stakeholders informed of the new oversight process, public meetings were held in the vicinity of each pilot plant. Public meetings were first held at the beginning of the pilot program, and then a series of Public Roundtable meetings were conducted at the end of the pilot program. These meetings were designed to both explain the new program, and then solicit feedback from the public on their views of the ROP.

Finally, a pilot program evaluation panel (PPEP) was established by the agency in accordance with Federal Advisory Committees Act (FACA) requirements to serve as an independent advisory committee to the agency. This panel was a cross-disciplinary group of managers and industry experts representing many different nuclear power interests, including the Union of Concerned Scientists, NEI, pilot plant licensee management, and the Illinois Department of Nuclear Safety, in addition to NRC Headquarters and regional management. The purpose of the PPEP was to independently evaluate the results of the pilot program and draw conclusions regarding required process changes and the readiness for initial implementation.

Culminating the feedback activities, the staff conducted a public lessons learned workshop from January 10-13, 2000. The purpose of the workshop was to bring internal and external stakeholders together to identify lessons learned and approaches to resolving key issues of concern. The workshop was successful in enabling the staff to achieve a good level of consensus on those issues requiring action prior to initial implementation, longer-term resolution, and continued monitoring during initial implementation.

On February 24, 2000, the staff issued Commission Paper SECY-00-0049 (Ref. 15) which forwarded to the Commission the results and lessons learned from the 6-month pilot program, results from internal and external stakeholder comments on the ROP, and the PPEP independent evaluation on the readiness of the new process for initial implementation. This paper also requested Commission approval to implement the ROP at all nuclear power plants. By SRM dated March 28, 2000, (Ref. 16) the Commission approved initial implementation of the revised ROP. Initial implementation of the new ROP for all commercial nuclear power plants commenced on April 2, 2000.

Although implemented at all nuclear power plants, the staff considered the first year of ROP implementation to be a time to collect additional insights and identify areas for program improvement. Similar to the 6-month pilot program, the staff employed many activities during ROP initial implementation to collect internal and external stakeholder feedback and comments and evaluate the new oversight process for lessons learned. As part of this effort, the staff developed a self-assessment program, described in Inspection Manual Chapter (IMC) 0307, "Reactor Oversight Process Self-Assessment Program," which utilizes objective measures and pre-determined criteria to monitor the performance of the ROP. Internal feedback and comments were obtained from Headquarters and regional staff while feedback and comments from external stakeholders, such as public interest groups, industry representatives, and state and local government agencies was also solicited.

The results and lessons learned from the first year of ROP implementation were documented by the staff in SECY-01-0114 (Ref. 17). As noted in this Commission paper, the staff will continue to periodically monitor and assess the effectiveness of the ROP to identify areas for improvement.

05.03 Regulatory Framework.

The foundation of the new ROP is the Regulatory Framework. The staff used a top-down, hierarchical approach to develop the concept for a new regulatory oversight framework that addresses the agency's regulatory principles. This approach started with a desired outcome, identified performance goals to achieve this outcome, and then identified specific objectives and information needs to meet each performance goal. The regulatory oversight framework developed by the staff using this approach is shown in Exhibit 2.

This framework starts at the highest level, with the NRC's overall mission to ensure that commercial nuclear power plants are operated in a manner that provides adequate protection of public health and safety. The staff then identified those aspects of licensee performance that are important to the mission and therefore merit regulatory oversight. The NRC Strategic Plan (Ref. 18) identifies the performance goals to be met for ensuring nuclear reactor safety and include the following:

- Maintain a low frequency of events that could lead to a nuclear reactor accident;

- Zero significant radiation exposures resulting from civilian nuclear reactors;

- No increase in the number of offsite releases of radioactive material from civilian nuclear reactors that exceed 10 CFR Part 20 limits; and

- No substantiated breakdown of physical protection that significantly weakens protection against radiological sabotage, or theft or diversion of special nuclear materials.

These performance goals reflect those areas of licensee performance for which the NRC has regulatory responsibility in support of the overall agency mission. These performance goals were represented in the framework structure as the strategic performance areas of Reactor Safety, Radiation Safety, and Safeguards, and formed the second level of the regulatory oversight framework.

With a risk-informed perspective, the staff then identified the most important elements in each of these strategic performance areas that form the foundation for meeting the overall agency mission. These elements were identified as the cornerstones of safety in the third level of the regulatory oversight framework structure. These cornerstones serve as the fundamental building blocks for the ROP, and acceptable licensee performance in these cornerstones should provide reasonable assurance that the overall mission of adequate protection of public health and safety is met.

The cornerstones of safety were chosen to: (1) limit the frequency of initiating events; (2) ensure the availability, reliability, and capability of mitigating systems; (3) ensure the integrity of the fuel cladding, reactor coolant system, and containment boundaries; (4) ensure the adequacy of the emergency preparedness functions; (5) protect the public from exposure to radioactive material releases; (6) protect nuclear plant workers from exposure to radiation; and (7) provide assurance that the physical protection system can protect against the design-basis threat of radiological sabotage.

Once the regulatory oversight framework was established, the staff developed defining principles that formed the strategy and rules for the further development of the details of the ROP. These defining principles established the relationship between elements of the oversight processes, such as enforcement and inspection, and include:

- There will be a risk-informed baseline inspection program that establishes the minimum regulatory interaction for all licensees.

- Thresholds can be set for licensee safety performance, below which increased NRC interaction (including enforcement) would be warranted.

- Adequate assurance of licensee performance at the cornerstone level requires assessment of both PIs and inspection findings.

- Both the PIs and results of inspections used to assess a cornerstone will have risk-informed thresholds.

- Crossing a PI threshold and an inspection threshold will have the same meaning with respect to safety significance and directly define the level of NRC involvement and action.

- The baseline inspection program will cover those risk-significant attributes of licensee performance not adequately covered by PIs.

- The baseline inspection program will also verify the accuracy of the PIs and provide for event response.

- Enforcement actions taken (e.g., the number of cited violations, the amount of a civil penalty) should not be an input into the assessment process. However, the issue that led to the enforcement action will continue to be considered in the assessment.

- Assessment process results might be used to modulate enforcement actions (although assessment results would not affect the determination of violation severity level).

- Guidelines will establish criteria for identifying and responding to unacceptable licensee performance.

It is important to note that the intent of these defining principles was to result in an oversight process that provides adequate margin in the assessment of licensee performance so that appropriate licensee and NRC actions are taken before unacceptable performance occurs.

05.04 Cornerstones of Safety.

The staff used a top-down, hierarchical, risk-informed approach for each cornerstone in an effort to:

- identify the objective and scope of the cornerstone;

- identify the desired results and important attributes of the cornerstone;

- identify what should be measured to ensure that the cornerstone objectives are met;

- determine which of the areas to be measured can be monitored adequately by PIs;

- determine whether inspection or other information sources are needed to supplement the PIs; and

- determine the thresholds of performance for each cornerstone, below which additional NRC actions would be taken.

Where possible, the staff sought to identify PIs as a means of measuring the performance of key attributes in each of the cornerstone areas. Where such a PI could not be identified, or where a PI was identified but was not sufficiently comprehensive, the staff identified a baseline inspection activity. The staff also identified the inspections necessary to verify the accuracy and completeness of the reported PI data. The results of applying the top-down, hierarchical approach to identify the PIs and baseline inspection necessary to meet the objectives of each cornerstone of safety are shown in Exhibits 3 through 10. Additional detail and discussion on the PIs and baseline inspection program for each cornerstone of safety can be found in IMC 308 Attachment 1 and 2.

For the reactor safety area, the cornerstones of safety are defined as follows:

Initiating Events. The objective of this cornerstone is to limit the frequency of those events that upset plant stability and challenge critical safety functions, during shutdown as well as power operations. If not properly mitigated and multiple barriers are breached, a reactor accident could result which would compromise the public health and safety. Licensees can reduce the likelihood of a reactor accident by maintaining a low frequency of these initiating events. Such events include reactor trips due to turbine trips, loss of feedwater, loss of off-site power, and other reactor transients.

Mitigating Systems. The objective of this cornerstone is to ensure the availability, reliability, and capability of systems that mitigate plant transients and the reactor accidents. Licensees reduce the likelihood of reactor accidents by enhancing the availability and reliability of mitigating systems. Mitigating systems include those systems associated with safety injection, residual heat removal, and their support systems, such as emergency AC power. This cornerstone includes mitigating systems that respond to both operating and shutdown events.

Barrier Integrity. The objective of this cornerstone is to ensure that physical barriers protect the public from radionuclide releases caused by accidents. Licensees can reduce the effects of reactor accidents or events if they do occur by maintaining the integrity of the barriers. The barriers are the fuel cladding, reactor coolant system boundary, and the containment.

Emergency Preparedness. The objective of this cornerstone is to ensure that actions taken by the emergency plan would provide protection of the public health and safety during a radiological emergency. Licensees can ensure that the emergency plan would be implemented correctly by drills and training. This would give reasonable assurance that the licensee can effectively protect the public health and safety in the event of a radiological emergency. This cornerstone does not include the off-site actions, which are covered by the Federal Emergency Management Agency.

For the reactor safety area to fail to meet the goal of adequate protection of public health and safety, an initiating event would have to occur, followed by failures in one or more mitigating systems, and ultimately failure of multiple barriers. At that stage, the emergency plan would be implemented as the last defense-in-depth measure for public protection.

For the radiation safety area, the cornerstones of safety are defined as follows:

Public Radiation Safety. The objective of this cornerstone is to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain as a result of routine civilian nuclear reactor operations. These releases include routine gaseous and liquid radioactive effluent discharges, the inadvertent release of solid contaminated materials, and the offsite transport of radioactive materials and wastes. Licensees can maintain public protection by meeting the applicable regulatory limits and "as low as is reasonably achievable" (ALARA) guidelines.

Occupational Radiation Safety. The objective of this cornerstone is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. This exposure could come from poorly controlled or uncontrolled radiation areas or radioactive material that unnecessarily exposes workers. Licensees can maintain occupational worker protection by meeting applicable regulatory limits and ALARA guidelines.

For safeguards, the cornerstone of safety is defined as follows:

Security. The objective of the security cornerstone is to provide assurance that the licensees's security system and material control and accounting program use a defense-in-depth approach and can protect against (1) the design basis threat of radiological sabotage from external and internal threats, and (2) the theft or loss of radiological materials.

Although the NRC is actively overseeing the security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary.

05.05 Cross-Cutting Issues, Substantive Cross-Cutting Issues, and Safety Culture Oversight.

In addition to identifying the seven cornerstones of safety, the staff also identified certain aspects of licensee performance that were seen as "cross-cutting" and potentially impacting more than one cornerstone. Aspects of licensee performance such as human performance, the establishment of a safety conscious work environment (SCWE), and the effectiveness of licensee problem identification and resolution programs, although not identified as specific cornerstones, are still important to meeting the agency safety mission. The staff concluded that these items generally manifest themselves as the root causes of performance problems. Adequate licensee performance in these cross-cutting areas will be assessed either explicitly in each cornerstone area or will be inferred through cornerstone performance results from both PIs and inspection results.

These cross-cutting issues are discussed below to characterize their significance and the means by which they were addressed during the cornerstone development process and subsequently in the June 2006 revision to the Reactor Oversight Process (ROP) to more fully address safety culture.

As part of the development activities for the June 2006 ROP revision, the staff adopted the International Atomic Energy Agency's International Nuclear Safety Advisory Group's definition of safety culture which "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." Further, Regulatory Issue Summary 2006-13, "Information on the Changes Made to the Reactor Oversight Process to More Fully Address Safety Culture" describes the changes made to selected ROP inspection procedures and manual chapters and the assessment process to address safety culture. Each of the three cross-cutting areas contains cross-cutting area components that are attributes of safety culture. The nine cross-cutting area components are fully described in IMC 0305.

a. Cross-Cutting Areas

1. Human Performance

By the nature of the design of nuclear power plants and the role of plant personnel in maintenance, testing, and operation, human performance plays an important role in normal, off-normal, and emergency operations. Following the accident at Three Mile Island, Unit 2 (TMI-2), the NRC implemented a number of programs that significantly improved the reliability of personnel performance and the safety of nuclear power plants by reducing the likelihood of core damage and containment failure. Detailed control room design reviews resulted in substantial improvements to the human engineering design of control rooms, as well as to control stations and panels outside the main control room. Emergency operating procedures were modified to include symptom-oriented mitigation strategies and were refined to be more useable, reducing errors in their implementation. Training programs for licensed operators, and later for other important plant personnel, were modified such that job-task analyses were performed which formed the basis for the development of learning objectives, training materials and approaches, objective-specific testing, and appropriate program improvements based on feedback from personnel performance in the field. Other policies and programs implemented by the NRC improved staffing, overtime controls, and fitness-for-duty of plant personnel. Still others improved security and safeguards operations, emergency planning and response, and health physics controls (both occupational and public). Broad-reaching verification and validation efforts were conducted to ensure the proper implementation of the programs. Together, these programs have significantly improved human performance.

Risk-informed, performance-based regulation will, at least in part, involve a shift in the NRC role from improving human reliability to one of monitoring human reliability. Past efforts were appropriately pro-active (rather than performance based) because the accident at TMI-2 had clearly illustrated the serious deficiencies in programs to support effective and safe human performance. The success of the human performance improvement programs allows the NRC to now take a more performance-based approach to regulatory oversight of human performance. Thus, if plant performance is acceptable (as monitored through risk-informed inspections and PIs), then the performance of plant personnel is assumed to be acceptable as well. That is, if risk-informed inspection (for example, maintenance rule verification inspections, configuration control inspections, and other inspections as described for each cornerstone) and plant PIs for each cornerstone (such as scrams and unplanned power changes for the initiating events cornerstone and safety system unavailability for the mitigating systems cornerstone) together indicate that plant performance is meeting the cornerstone objectives, then those findings also provide an indication of the acceptability of the associated human activities. This relationship between plant and human performance is assumed to be especially strong with regard to the broad range of normal operations, including maintenance and testing activities during power and shutdown operations. Routine baseline inspections of licensee problem identification and resolution programs will be conducted to ensure that human performance (and those factors such as training, procedures, and the like that influence human performance) is specifically and appropriately investigated through licensees' root cause analyses and corrective action programs, including the investigation of potential common cause failures caused by human actions.

Post-initiator operator actions are far less frequent than pre-initiator human activities that influence the latent capability of plant equipment. While initial and requalification examinations provide a predictive measure of operator performance during off-normal and emergency operations, follow-up inspections of risk-significant events will provide a more direct indication of the adequacy of post-initiator human performance. In addition, performance measures from emergency response exercises, and those associated with security and occupational exposure, will provide another means for the NRC to ensure that human reliability is being maintained appropriately.

2. Safety Conscious Work Environment

A SCWE is defined as an environment in which employees feel free to raise safety concerns, both to their management and to the NRC, without fear of retaliation and where such concerns are promptly reviewed, given the proper priority based on their potential safety significance, and appropriately resolved with timely feedback to employees [by licensee management]. SCWE is an important attribute of safety culture. In general, management commitment to safety will promote a SCWE. Possible indications of an "unhealthy" safety culture include a high number of allegations, a reticence of licensee employees to use internal processes to raise safety concerns, and a high corrective maintenance backlog.

SCWE is a cross-cutting area since an unhealthy SCWE can affect performance in any of the cornerstone areas. For example, weaknesses in an environment for raising concerns or for not preventing, detecting, and mitigating perceptions of retaliation and reluctance of licensee staff to raise nuclear safety concerns can result in deficiencies going unresolved, which could complicate plant response to a subsequent event (mitigating systems or barriers cornerstone).

The importance of a SCWE is similar to, if not integral with, the role of licensee problem identification and corrective action processes. As with the problem identification and corrective action cross-cutting issue, an assumption was made regarding the role of a SCWE in NRC assessments of licensee performance. Specifically, if a licensee had a poor SCWE, problems and events would continue to occur at that facility to the point where either they would result in exceeding thresholds for various PIs, or they would be surfaced during NRC baseline inspection activities, or both. Additionally, because inspection of licensee problem identification and corrective action programs will be included in the baseline inspection program (through IP 71152, "Identification and Resolution of Problems"), some indirect assurance will be gained as to the health of a licensee's safety culture. In short, no separate and distinct assessment of licensee safety culture is needed because it is subsumed by either the PI's or baseline inspection activities.

3. Problem Identification and Resolution Programs

Defining and implementing an effective problem identification and resolution program is a key element underlying licensee performance in each cornerstone area. A fundamental goal of the NRC's reactor inspection and assessment process is to establish confidence that each licensee is detecting and correcting problems in a manner that limits the risk to members of the public. The NRC expects licensees to be technically and organizationally self-sufficient in this regard. Ineffective problem identification and resolution programs, including poor conduct of root cause analysis of self-identified or self-revealing issues, has been a common theme among problem plants in the past. The scope of problem identification and resolution programs includes processes for self-assessment, root cause analysis, safety committees, operating experience feedback, and corrective action.

With regard to licensee problem identification and resolution effectiveness, there are several areas that are not specifically evaluated by either the individual cornerstone PIs or the individual risk-informed inspections. As such, additional focused inspection is needed to evaluate licensee performance as it relates to this cross-cutting issue. Specifically, baseline inspection of licensee corrective action programs is necessary for the NRC to:

- conduct reviews of precursors to events which occur relatively infrequently but could have significant consequences;

- independently identify potentially "generic" concerns that a licensee may have missed, including specific problems involving safety equipment, procedure development, design control, etc.;

- have assurance that licensees adequately address potential "common cause" equipment failure concerns, identified either by internal events and issues or by receipt of operating experience feedback from other licensees, vendors, etc.

Also these inspections provide the NRC with early warning of potential performance issues that could result in crossing thresholds in the Action Matrix and help the NRC gauge supplemental response should future Action Matrix thresholds be crossed. The inspections provide insights into whether licensees have established a SCWE and allow for follow-up of previously identified compliance issues (e.g., non-cited violations). The inspections also provide additional information that can be used in the assessment process, beyond that which is provided by the SDP.

b. Substantive Cross-Cutting Issues

The NRC has developed criteria for notifying the licensee when a substantive cross-cutting issue exists at a particular site. The purpose of identifying a substantive cross-cutting issue at the mid-cycle and end-of-cycle review meetings is to inform the licensee on the docket that the NRC has a significant level of concern with the licensee's performance in the cross-cutting area. The June 2006 revision modified the decision making process for determining a substantive cross-cutting issue, as well as the possible NRC actions if a substantive cross-cutting issue is not addressed in a timely manner. The specific guidance on implementing the assessment of substantive cross-cutting issues is described in IMC 0305.

c. Safety Culture Oversight

In addition to the nine cross-cutting area components that fall within the three cross-cutting areas (see a. above), which are assessed during the baseline inspection program and assessment process, the staff identified four additional safety culture components (which do not fall within the three cross-cutting areas). This new total, 13 safety culture components are described in IMC 0305. All 13 safety culture components are considered by inspectors to provide insight into the adequacy of the licensee's root cause, extent of condition, and safety culture evaluations during a baseline inspection (IP 71152, "Identification and Resolution of Problems"), special reactive inspections (IPs 71153, "Event Follow-up," 93800, "Augmented Inspection Team," and 93812, "Special Inspection Team") and supplemental inspections (IPs 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," 95002, "Supplemental Inspection for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," and 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input.").

Specifically during the problem identification and resolution inspection the inspectors will consider all of the safety culture components while evaluating the adequacy of the licensee actions. During reactive inspections potential contributing causes related to all of the safety culture components are considered as part of the efforts to fully understand the circumstances surrounding a plant event and its probable causes. In a similar manner, during the supplemental inspection IP 95001 the inspectors will verify that the licensee's root cause, extent of condition, and extent of cause evaluations appropriately considered all of the safety culture components. During IP 95002 the inspectors will independently determine whether one or more of any of the safety culture components caused or significantly contributed to the risk-significant performance issue. For licensees whose performance has degraded to the point they are in the multiple/repetitive degraded cornerstone column, the staff will perform IP 95003 where all of the safety culture components will be used as the staff evaluates the licensee's independent third-party safety culture assessment, and when the staff performs its own independent assessment of the licensee's safety culture.

05.06 <u>Risk-Informed Scale</u>.

In developing the new performance assessment process, one of the tasks was to establish risk-informed 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 consideration of risk and regulatory response to different levels of licensee performance. The approach was intended to be consistent with other NRC risk-informed regulatory applications and policies as well as consistent with regulatory requirements and limits. The primary attributes of the original concept were:

- the scheme should include multiple levels with clearly defined thresholds to allow unambiguous observation and assessment of declining (or improving) performance;

- the thresholds should be risk informed to the extent practical, but should accommodate defense-in-depth and indications based on existing regulatory requirements and safety analyses;

- the risk implications and regulatory actions associated with each performance band and associated threshold should be consistent with other NRC risk applications, and based on existing criteria where possible (e.g., Regulatory Guide [RG] 1.174 [Ref. 19]);

- the scheme should provide for consistency of risk-informed indications of performance which are based on existing regulatory requirements and safety analyses to the extent practical;

- the scheme should be capable of accounting for performance indicated by risk-informed inspection findings;

- thresholds that cannot be risk-informed should be set at levels that will result in the level of regulatory response necessary to address the finding;

- thresholds should provide sufficient differential to allow meaningful differentiation in performance and limit false positives (e.g., allow an order of magnitude in the risk differential between thresholds);

- sufficient margin should exist between nominal performance bands to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement thresholds; and sufficient margin should exist between thresholds that signify initial declining performance to allow for both NRC and licensee diagnostic and corrective actions to be effective before licensee performance becomes unacceptable;

- each individual PI should have its own performance thresholds;

- where appropriate plant-specific design differences should be accommodated; and

The basis for establishing these performance thresholds was RG 1.174, which brings in the Regulatory Analysis Guidelines (Ref. 20), and the Safety Goal Policy Statement (Ref. 21). The metrics that have been adopted in RG 1.174 for the characterization of risk are core damage frequency (CDF) and large early release frequency (LERF). These are essentially surrogates for health effects, which are the principal metrics in the Safety Goal Policy Statement, and, in addition, they are consistent with the metrics used in the Regulatory Analysis Guidelines. In RG 1.174, acceptance guidelines were established for assessing changes to the licensing basis of a plant. Acceptance is predicated on increases in CDF and LERF implied by the change to the licensing basis being small.

The philosophy behind the establishment of the thresholds on PIs and inspection findings was essentially to assume that an increase in PI values or conditions indicated by the finding, would, if their root causes were uncorrected, be equivalent to accepting a de facto increase in the CDF and LERF metrics. This is clearer for the PIs than it is for the inspection findings, which may relate to a time limited undesired condition. For such cases, the model used here is that the event is indicative of an underlying performance issue that, if uncorrected, would be expected to result in similar occurrences with the same frequency.

Therefore, the challenge was how to calculate the impact of changes in PI values and inspection findings on these metrics. Since PIs correspond (at least in some approximate sense) to parameters of PRA models, it was relatively straightforward to make the connection between changes in PI values to changes in risk. The thresholds were established by taking a set of PRA models, and varying the parameter that corresponded to the PI until the change in CDF became 10^{-5} or 10^{-4}/yr, and these values were chosen as the thresholds for the White/Yellow and Yellow/Red thresholds. Therefore, the risk significance of an inspection finding should be measured in the same way. When the impact of the finding can be characterized in terms of the unavailability of an SSC for some specified duration, then the SDP gives an estimate of the change in CDF.

As shown in Exhibit 12, a conceptual model was developed to incorporate the attributes listed above. This model was used as the basis for developing the thresholds and performance bands for PIs and inspection findings and their general performance characteristics are as discussed below:

- The licensee response band is characterized by acceptable performance in which cornerstone objectives are fully met; nominal risk with nominal deviation from expected performance. This performance band is designated as the Green band. Performance problems would not be of sufficient significance that escalated NRC engagement would occur. Licensees would have maximum flexibility to "manage" corrective action initiatives.

- The increased regulatory response band would be entered when licensee performance is outside the normal performance range, but would still represent an acceptable level of performance. This performance band is designated as the White band. Cornerstone objectives met with minimal reduction in safety margin; outside bounds of nominal performance; within Technical Specification Limits. Degradation in performance in this band is typified by changes in risk of up to 10^{-5} ΔCDF or 10^{-6} ΔLERF associated with either PIs or inspection findings. The CDF and LERF threshold characteristics were selected to be consistent with RG 1.174 applications.

- The required regulatory response band involves a decline in licensee performance that is still acceptable with cornerstone objectives met, but with significant reduction in safety margin; Technical Specification limits reached or exceeded. This performance band is also designated as the Yellow band. Degradation in performance in this band is typified by changes in risk of up to 10^{-4} ΔCDF or 10^{-5} ΔLERF associated with either PIs or inspection findings. These threshold characteristics and required regulatory response are also selected to be consistent with risk-informed regulatory applications and mandatory actions for regulatory compliance.

- The Red band is typified by changes in performance that are indicative of changes in risk greater than 10^{-4} ΔCDF or 10^{-5} ΔLERF associated with either PIs or inspection findings. Plant performance represents an unacceptable loss of safety margin. It should be noted that should licensee's performance result in a PI reaching the Red Band, margin would still exist before an undue risk to public health and safety would be presented.

This conceptual model was also applied to the determination of overall plant performance through the assessment process Action Matrix. As described in IMC 308 Attachment 4, the thresholds for each column of the Action Matrix were established using the conceptual model of Exhibit 12 to indicate declining licensee performance of a more pervasive and systemic nature as you proceed from the left-most column across the Action Matrix. However, there were fundamental differences between applying the concept of performance bands to individual assessment inputs (PIs and inspection findings) and to overall plant performance (Action Matrix).

First and foremost is that while an individual performance issue in the Yellow band may indicate a significant safety concern regarding a specific aspect or area of licensee performance, this single issue represents only a minimal reduction in overall plant safety. This is the result of the defense-in-depth concept used in the design of plants, and causes the columns of the Action Matrix to not align directly with the performance bands of Exhibit 12.

The second major difference is that the Action Matrix is composed of five performance columns, while the conceptual model only has four performance bands. This was necessary to reflect the fact that a Red input may in some cases, but not always, reflect an overall level of licensee performance that is unacceptable. Just as was the case for the Yellow band discussed above, while an individual Red input may indicate a performance issue that is significantly degraded, overall plant performance may not be unacceptable due to the defense-in-depth design of the plants. Therefore to reflect this situation, two columns were created to describe the NRC's response to both an acceptable and unacceptable overall level of performance due to a Red assessment input.

05.07 Commission Commitments

During the development of the ROP, the Commission provided significant direction to the staff regarding certain attributes that the ROP should address. These items helped form the foundation of the ROP, and establish the basis for many important features of the ROP. These items, for the most part, come from Commission SRMs that were issued in response to many of the papers written and briefs conducted during ROP development. A summary of the more significant items that influenced the development of the ROP and subsequent Commission direction related to safety culture oversight follows:

a. SRM for SECY-98-045, dated June 30, 1998 (Ref. 8)

    - While the enforcement program is a valuable regulatory tool, the Commission does not desire that enforcement be used as a "driving force" of the assessment activities.

b. SRM for SECY-99-007 and SECY-99-007A, dated June 18, 1999 (Ref. 12)

    - The staff should consider ways to ensure that the assessment process is sufficiently robust to address programmatic breakdowns (e.g., breakdown of a corrective actions program or aspects of a particular quality assurance program) which are different from issues involving many minor findings. Consistent with this approach, and the overall direction of the changes to the inspection, assessment, and enforcement programs, the staff should not continue to evaluate the feasibility of designing a system to analyze the risk significance of numerous problems of lower safety significance, which in the aggregate could be significant.

- The Commission should be briefed annually regardless of whether any plants are identified for agency-level action.

- The staff should provide licensees (and the public) with fourth quarter assessments prior to the annual Commission meeting to aid licensees' efforts to address NRC concerns, to provide due process, and to ensure against "surprises" coming out at the meeting.

- The staff should consider how it will address licensee-identified issues so as to not discourage licensees from having an aggressive problem-identification process.

c. SRM for SECY-00-0049, dated May 17, 2000 (Ref. 22)

- The staff should minimize deviations from the Action Matrix, clearly document the basis for the deviations, and clearly explain the basis for deviations to all stakeholders.

- NRR and regional management should take steps to assure that inspector observations are placed in an appropriate context and do not undermine the overall effort to put inspection and enforcement efforts on a more objective and consistent foundation.

- The staff should show that cross-cutting issues they identify have a clear and strong link to significant inspection findings or degraded PIs before the staff attempts to take action on programmatic concerns.

d. SRM for SECY-04-0020, dated March 29, 2004 (Ref. 23)

- The staff should develop a separate process to address how security-related inspection findings and performance indicators would be considered when determining appropriate agency response. In developing a separate but parallel ROP process for physical protection, the staff should engage the industry through the existing Security Working Group arrangement, seeking clarification from the Security Steering Committee on emerging issues and consult with the Commission, as appropriate, when warranted.

e. SRM for SECY-04-0111, dated August 30, 2004 (Ref. 24)

- The staff should enhance the ROP treatment of cross-cutting issues to more fully address safety culture.

- The staff should include as part of the inspection activities for plants in the degraded cornerstone column of the ROP action matrix, a determination of the need for a specific evaluation of the licensee's safety culture and develop a process for making the determination and conducting the evaluation.

f. SRM for SECY-05-0187, Dated December 21, 2005 (Ref. 25)

- The staff should 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.

- The staff should 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.

- The staff should document significant changes to the ROP addressing safety culture in the ROP guidance documents and/or basis documentation.

- The staff should ensure the resulting modifications to the ROP are consistent with the regulatory principles that guided the development of the ROP.

0308-06 ACRONYMS AND REFERENCES

06-01 Acronyms.

AARM	Agency Action Review Meeting
AEOD	Office for Analysis and Evaluation of Operational Data
ALARA	As Low As is Reasonably Achievable
ANS	Alert and Notification System
ASP	Accident Sequence Precursor
BC	Branch Chief
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCTV	Closed Circuit Television
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
ΔCDF	Change in Core Damage Frequency
CoC	Certificates of Compliance
DD	Division Director
DEP	Drill/Exercise Performance
DHR	Decay Heat Removal
DID	Defense-in-Depth
DOT	U.S. Department of Transportation
DR	Degradation Rating
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DRT	Double Room Term
EDO	Executive Director for Operations
EP	Emergency Preparedness
EPRI	Electric Power Research Institute
ERO	Emergency Response Organization

FACA	Federal Advisory Committees Act
FEMA	Federal Emergency Management Agency
FFD	Fitness-for-Duty
FMF	Fire Mitigation Frequency
FPRSSM	Fire Protection Risk Significance Screening Methodology
FRN	Federal Register Notice
HPCI	High-pressure Coolant Injection
HRA	High Radiation Area
IDS	Intrusion Detection Systems
IF	Ignition Frequency
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPE	Individual Plant Evaluations
IPEEE	Individual Plant Examination of External Events
IRAP	Integrated Review of the Assessment Processes
ISI	Inservice Inspection
ISLOCA	Interfacing System Loss-of-Coolant-Accident
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPCI	Low-pressure Coolant Injection
LPCS	Low-pressure Core Spray
MD	Management Directive
MOV	Motor-Operated Valve
MR	Maintenance Rule
MSPI	Mitigating System Performance Index
NCV	Non-Cited Violations
NEI	Nuclear Energy Institute
NOV	Notice of Violation
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
NUS	Nuclear Utilities Service
ODCM	Offsite Dose Calculation Manual
OE	Office of Enforcement
OGC	Office of General Counsel
OI	Office of Investigations
OSRE	Operational Security Response Evaluation
PA	Protected Area
PAR	Protective Action Recommendation
PI	Performance Indicator
PIM	Plant Issues Matrix
PI&R	Problem Identification and Resolution
PPEP	Pilot Program Evaluation Panel
PPR	Plant Performance Review
PPSDP	Physical Protection Significance Determination Process

PRA	Probabilistic Risk Assessment
PS	Planning Standards
PWR	Pressurized-water Reactor
RA	Regional Administrator
RCIC	Reactor Core Isolation Coolant
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RES	Office of Nuclear Regulatory Research
RETS	Radiological Effluent Technical Specifications
RG	Regulatory Guide
RI	Resident Inspector
ROP	Reactor Oversight Process
RP	Radiation Protection
RSPS	Risk Significant Planning Standard
SALP	Systematic Assessments of Licensee Performance
SAMG	Severe Accident Management Guidelines
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generated Tube Rupture
SI	Special Inspection
SMM	Senior Management Meeting
SOC	Statements of Consideration
SRA	Senior Reactor Analyst
SRI	Senior Resident Inspector
SRM	Staff Requirements Memorandum
SRT	Single Room Term
SSA	Safety System Actuation
SSCs	Structures, Systems, and Components
SSD	Safe Shutdown
SSF	Safety System Failure
TEDE	Total Effective Dose Equivalent
TMI-2	Three Mile Island, Unit 2
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VHRA	Very High Radiation Area

06-02 References

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2. A. Andersen, "Recommendations to Improve the Senior Management Meeting Process," December 30, 1996

3. "Staff Action Plan to Improve the Senior Management Meeting Process," SECY-97-072, April 2, 1997

4. "Staff Requirements—Briefing on Staff Response to Arthur Andersen Study Recommendations," Staff Requirements Memorandum M970424B, June 24, 1997

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6. "Staff Requirements—SECY-97-122—Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors," Staff Requirements Memorandum 9700238, August 19, 1997

7. "Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors," SECY-98-045, March 9, 1998

8. "Staff Requirements - SECY-98-045 - Status of The Integrated Review of The NRC Assessment Process For Operating Commercial Nuclear Reactors," Staff Requirements Memorandum, June 30, 1998

9. NEI, "A New Regulatory Oversight Process," July 27, 1998

10. "Recommendations for Reactor Oversight Process Improvements," SECY-99-007, January 8, 1999

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12. "Staff Requirements - SECY-99-007 - Recommendations For Reactor Oversight Process Improvements and SECY-99-007a - Recommendations For Reactor Oversight Process Improvements (Follow-up to SECY-99-007)," Staff Requirements Memorandum, June 18, 1999

13. S. J. Collins, "Pilot Program for the New Regulatory Oversight Process," Memorandum, May 20, 1999

14. USNRC, "Public Comment on the Pilot Program for the New Regulatory Oversight Program," Federal Register, Vol. 64, p. 40394 (64 FR 40394), July 26, 1999

15. “Results of the Revised Reactor Oversight Process Pilot Program," SECY-00-0049, February 24, 2000

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19. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998

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21. USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986

22. "Staff Requirements - SECY-00-0049 - Results of the Revised Reactor Oversight Process Pilot Program (Part 2)," Staff Requirements Memorandum, May 17, 2000

23. "Staff Requirements - SECY-04-0020 - Treatment of Physical Protection Under the Reactor Oversight Process," Staff Requirements Memorandum, March 29, 2004 (Non-Publicly Available)

24. "Staff Requirements - SECY-04-0111 - Recommend Staff Actions Regarding Agency Guidance In The Areas of Safety Conscious Work Environment and Safety Culture," Staff Requirements Memorandum, August 30, 2004

25. "Staff Requirements - SECY-05-0187 - Status of Safety Culture Initiatives and Schedule for Near term Deliverables," Staff Requirements Memorandum, December 21, 2005

END

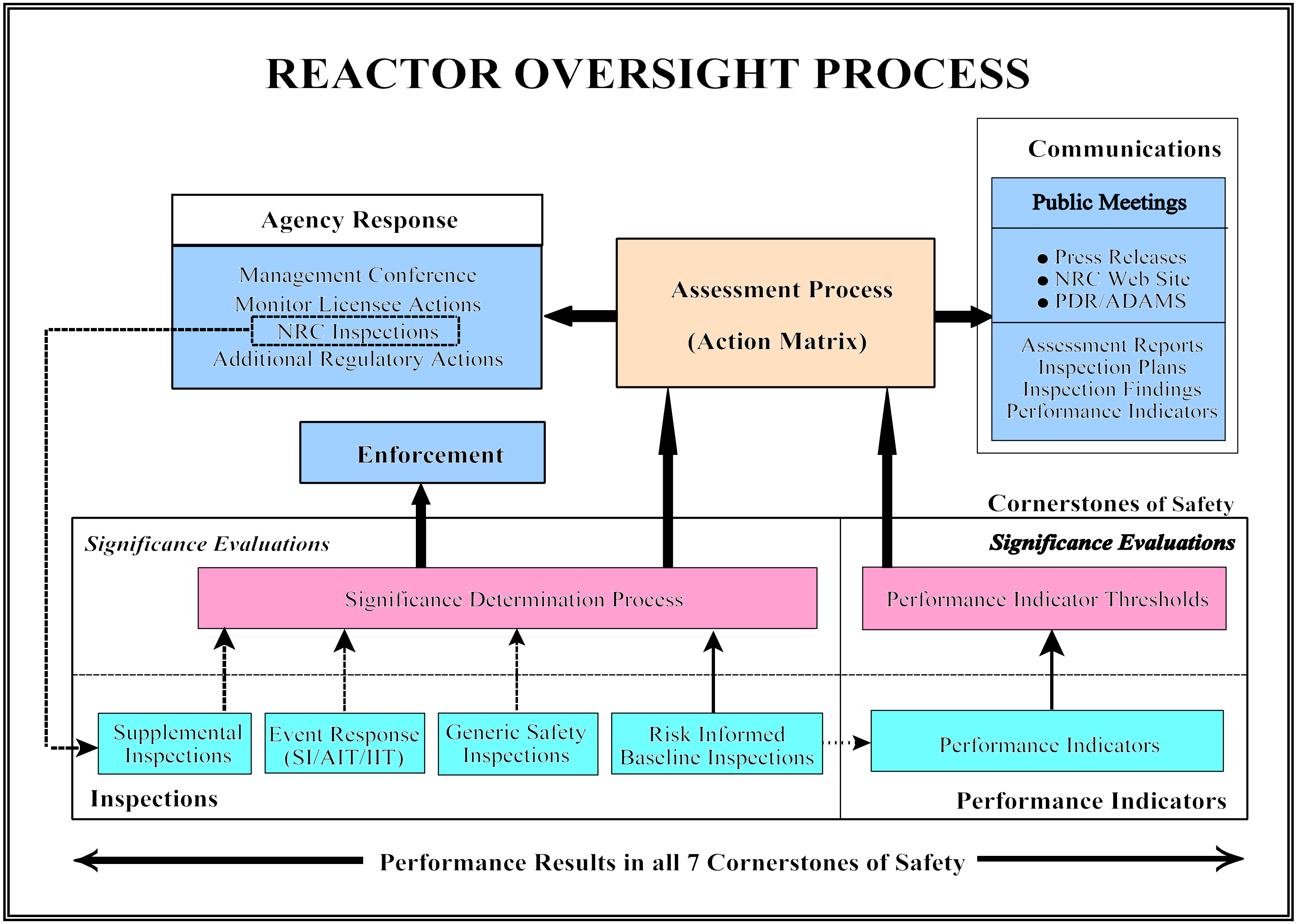
Exhibits:
1. Reactor Oversight Process
2. Regulatory Framework
3. Initiating Events Cornerstone Diagram
4. Mitigating Systems Cornerstone Diagram
5. Barrier Integrity Cornerstone Diagram - Fuel Cladding
6. Barrier Integrity Cornerstone Diagram - RCS
7. Barrier Integrity Cornerstone Diagram - Containment
8. Emergency Preparedness Cornerstone Diagram
9. Occupational Radiation Safety Cornerstone Diagram
10. Public Radiation Safety Cornerstone Diagram
11. Physical Protection Cornerstone Diagram
12. Risk Scale Conceptual Model

Attachments:
1. Performance Indicators
2. Inspection Program
3. Significance Determination Process

Appendices:
- A. Technical Basis for At Power Significance Determination Process
- B. Technical Basis for Emergency Preparedness Significance Determination Process

C. Technical Basis for Occupational Radiation Safety Significance Determination Process
D. Technical Basis for Public Radiation Safety Significance Determination Process
E. Technical Basis for Physical Protection Significance Determination Process (Reserved)
F. Technical Basis for Fire Protection Significance Determination Process
G. Technical Basis for Shutdown Operations Significance Determination Process
H. Technical Basis for Containment Integrity Significance Determination Process
I. Technical Basis for Operator Requalification Human Performance Significance Determination Process
J. Technical Basis for Steam Generator Tube Integrity Findings Significance Determination Process
K. Technical Basis for Maintenance Risk Assessment and Risk Management SDP

4. Assessment
5. Enforcement Policy


REACTOR OVERSIGHT PROCESS
Communications
Public Meetings
● Press Releases
● NRC Web Site
● PDR/ADAMS
Assessment Reports
Inspection Plans
Inspection Findings
Performance Indicators
Agency Response
Management Conference
Monitor Licensee Actions
NRC Inspections
Additional Regulatory Actions
Assessment Process
(Action Matrix)
Enforcement
Cornerstones of Safety
Significance Evaluations
Significance Evaluations
Significance Determination Process
Performance Indicator Thresholds
Supplemental Inspections
Event Response (SI/AIT/IIT)
Generic Safety Inspections
Risk Informed Baseline Inspections
Performance Indicators
Inspections
Performance Indicators
Performance Results in all 7 Cornerstones of Safety

REGULATORY FRAMEWORK

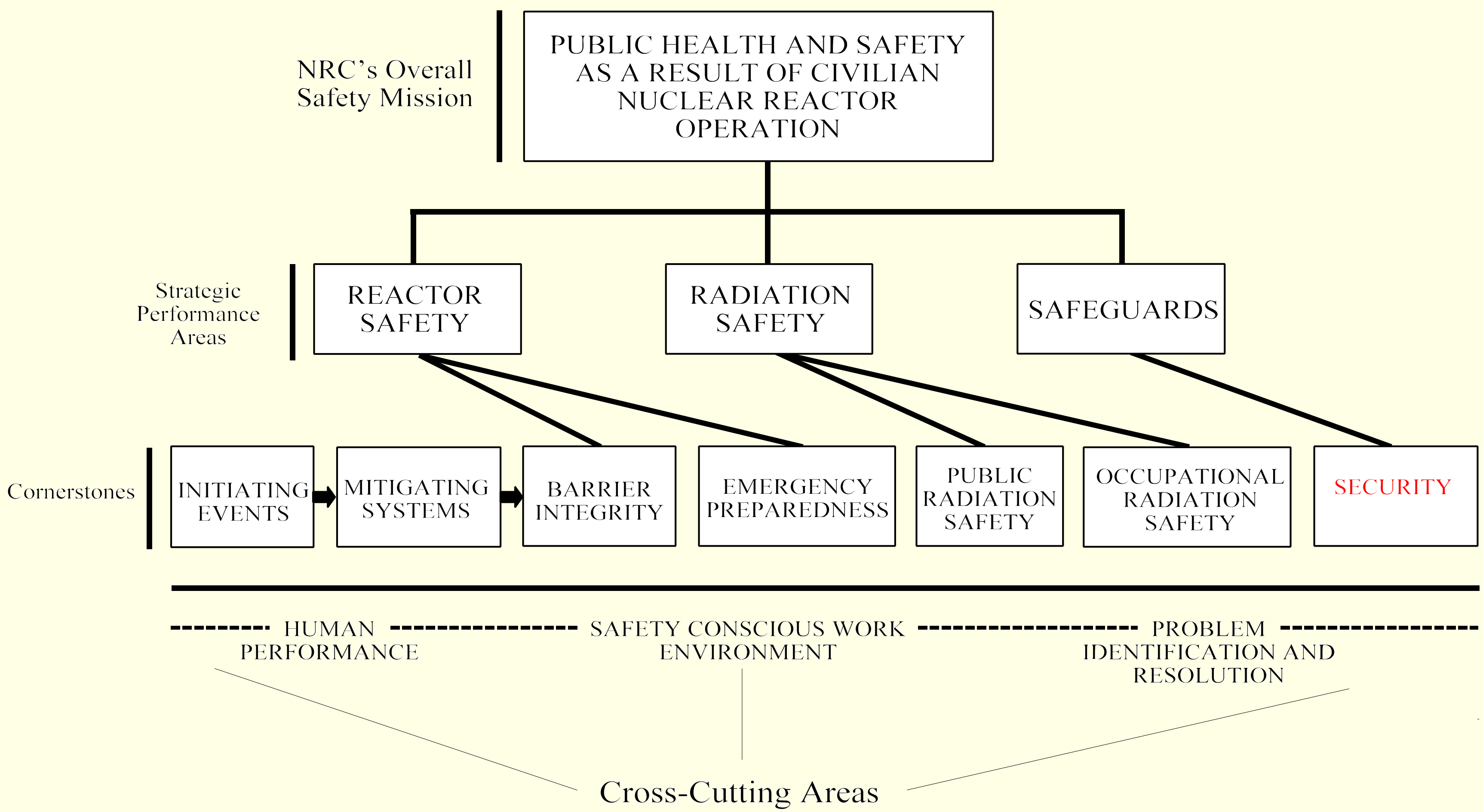
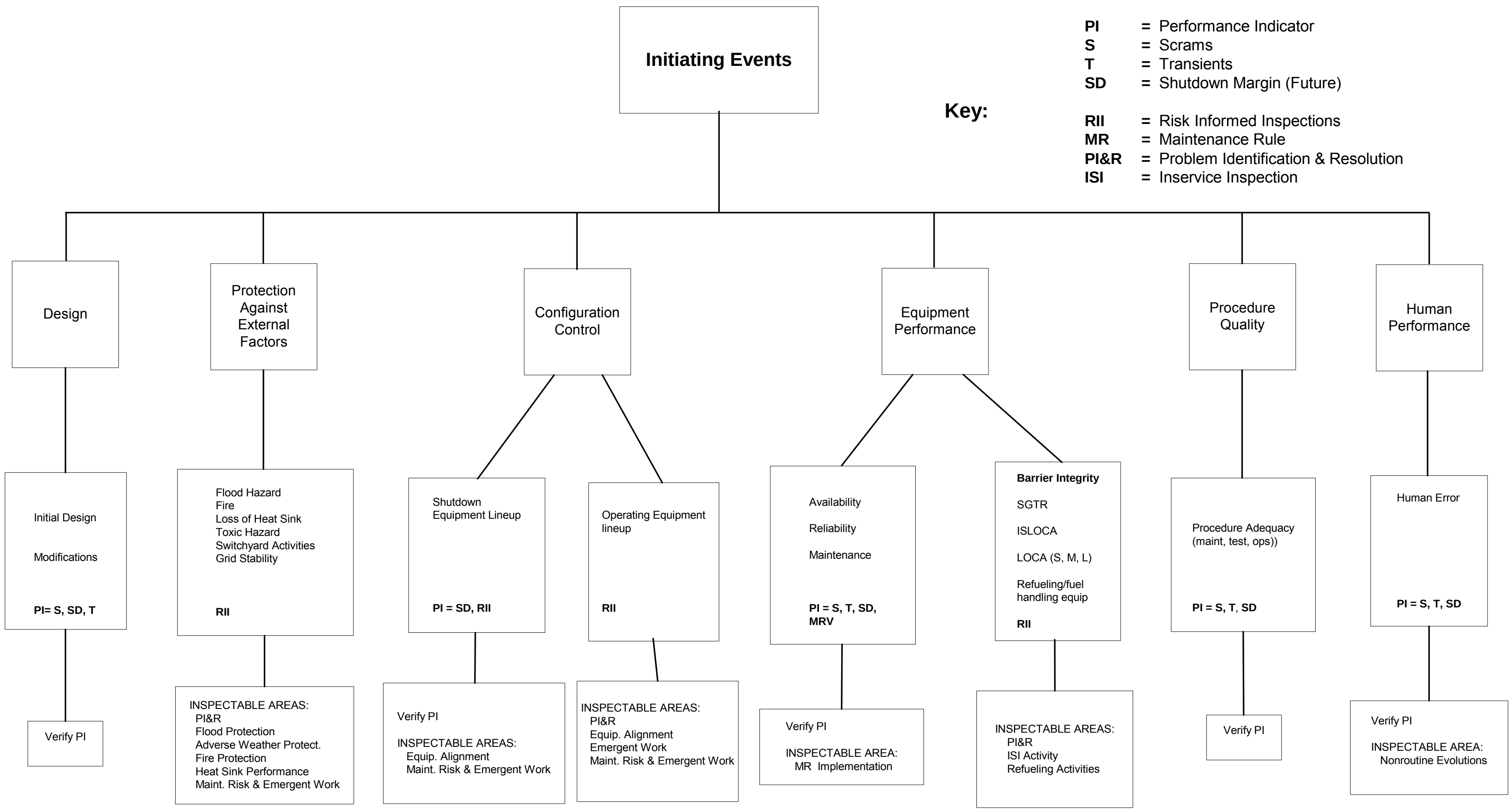


EXHIBIT 2

EXHIBIT 3



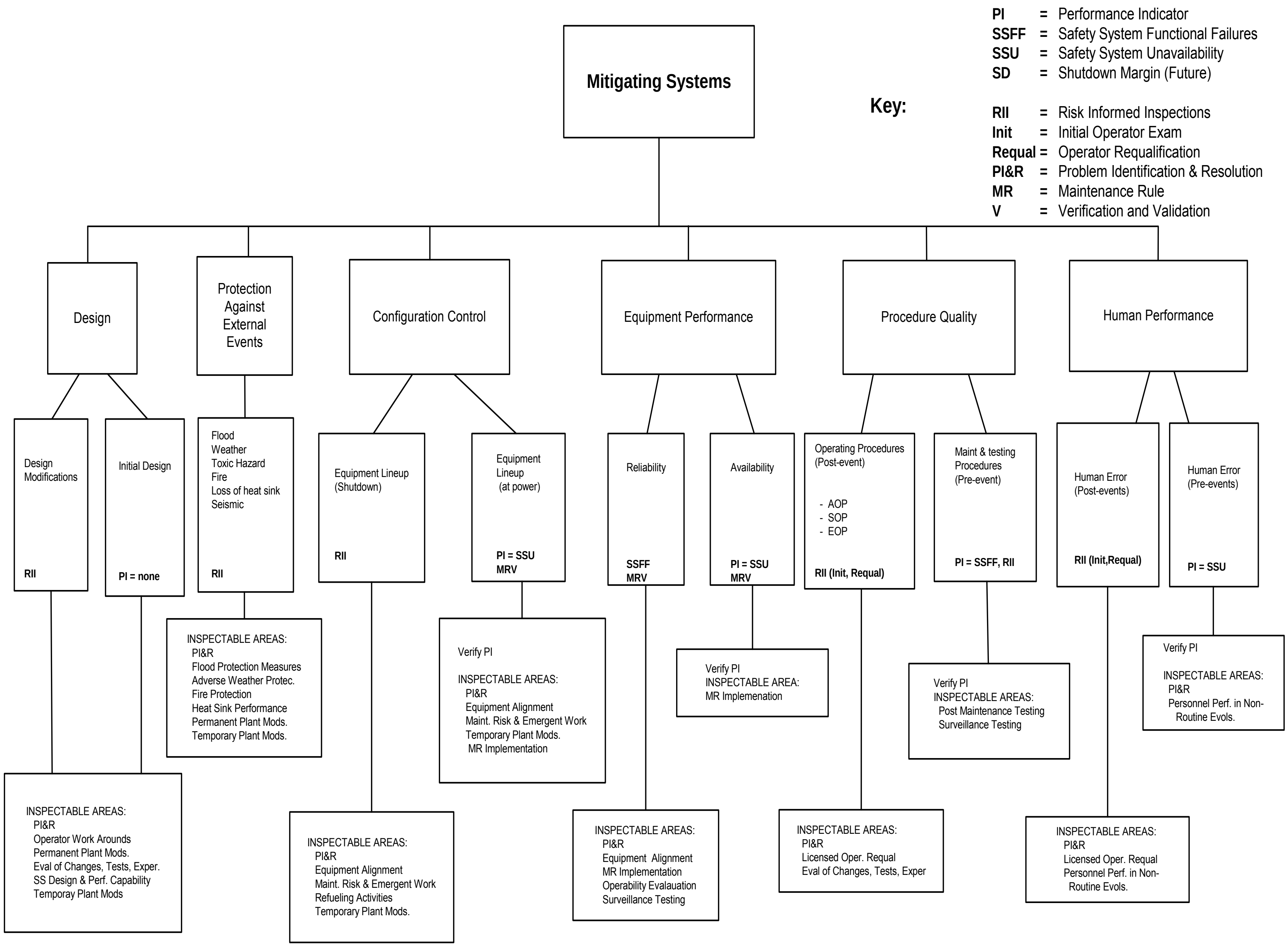
Key:

PI = Performance Indicator
S = Scrams
T = Transients
SD = Shutdown Margin (Future)

RII = Risk Informed Inspections
MR = Maintenance Rule
PI&R = Problem Identification & Resolution
ISI = Inservice Inspection

March 1, 2000

EXHIBIT 4



January 30, 2001

EXHIBIT 5

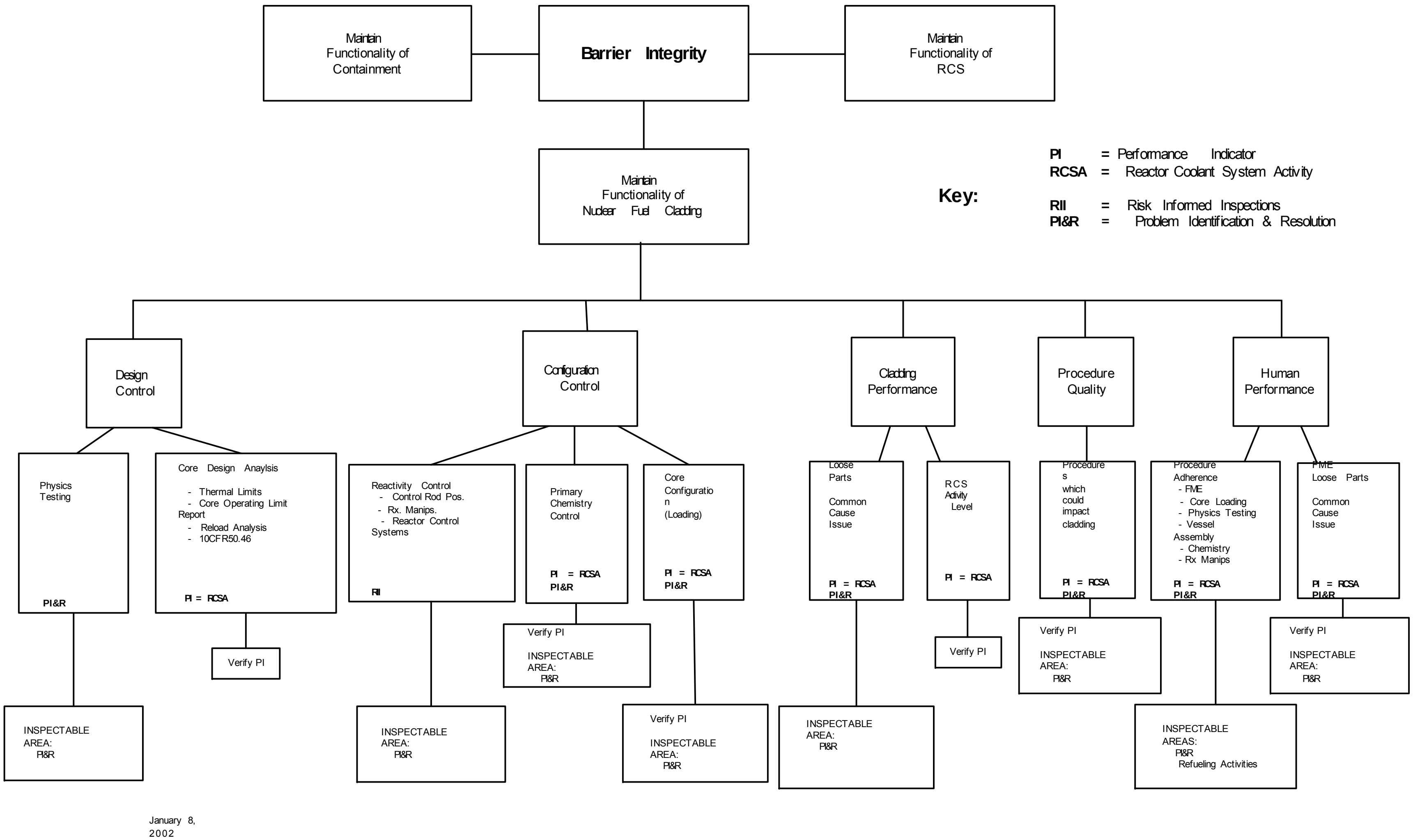
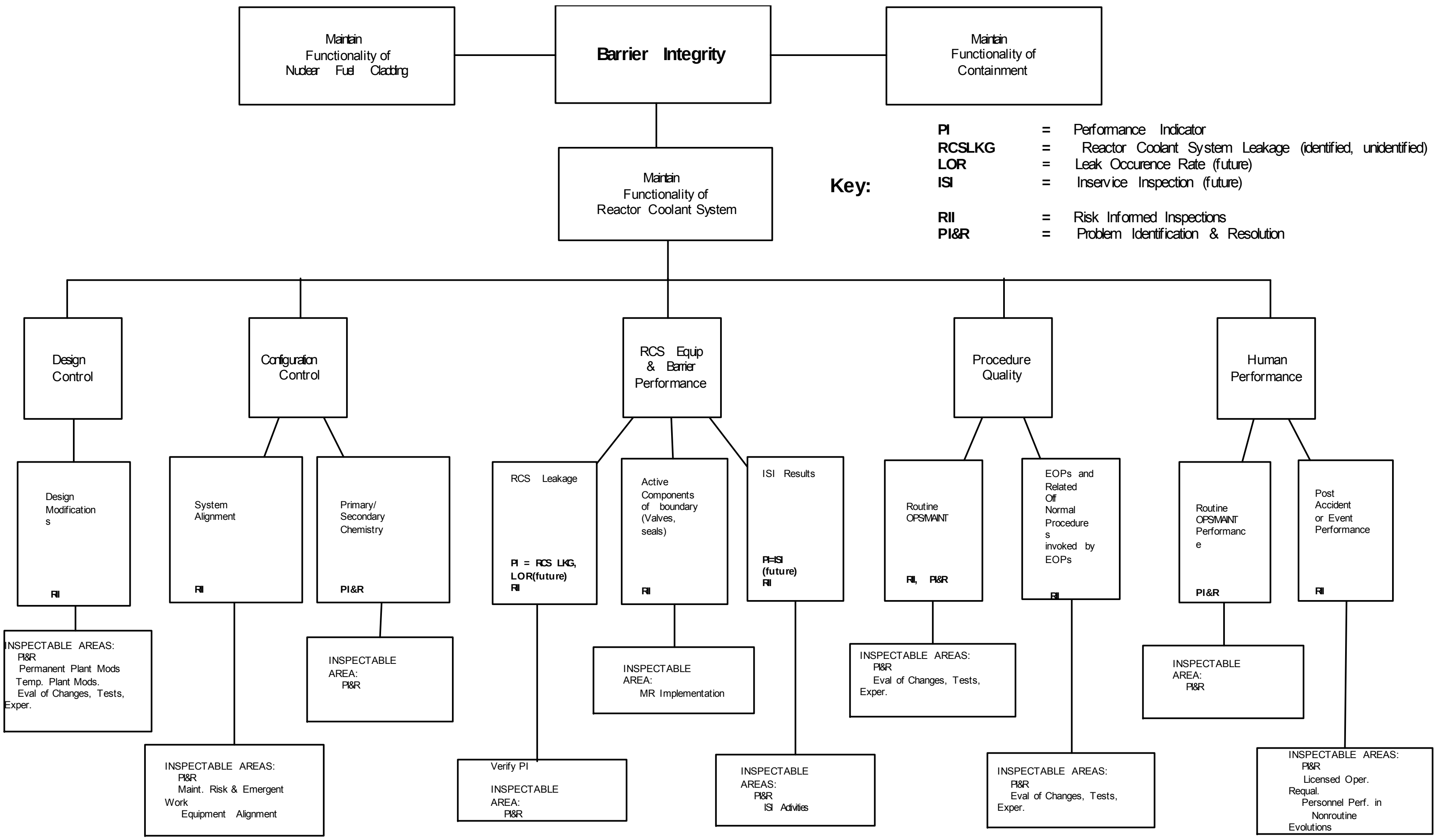

Maintain Functionality of Containment
Barrier Integrity
Maintain Functionality of RCS
Maintain Functionality of Nuclear Fuel Cladding
Key:
PI = Performance Indicator
RCSA = Reactor Coolant System Activity
RII = Risk Informed Inspections
PI&R = Problem Identification & Resolution
Design Control
Configuration Control
Cladding Performance
Procedure Quality
Human Performance
Physics Testing
PI&R
Core Design Anaylsis
- Thermal Limits
- Core Operating Limit Report
- Reload Analysis
- 10CFR50.46
PI = RCSA
Verify PI
INSPECTABLE AREA: PI&R
Reactivity Control
- Control Rod Pos.
- Rx. Manips.
- Reactor Control Systems
RII
INSPECTABLE AREA: PI&R
Primary Chemistry Control
PI = RCSA
PI&R
Verify PI
INSPECTABLE AREA: PI&R
Core Configuration (Loading)
PI = RCSA
PI&R
Verify PI
INSPECTABLE AREA: PI&R
Loose Parts
Common Cause Issue
PI = RCSA
PI&R
INSPECTABLE AREA: PI&R
RCS Activity Level
PI = RCSA
Verify PI
Procedures which could impact cladding
PI = RCSA
PI&R
Verify PI
INSPECTABLE AREA: PI&R
Procedure Adherence
- FME
- Core Loading
- Physics Testing
- Vessel Assembly
- Chemistry
- Rx Manips
PI = RCSA
PI&R
INSPECTABLE AREAS: PI&R
Refueling Activities
FME
Loose Parts
Common Cause Issue
PI = RCSA
PI&R
Verify PI
INSPECTABLE AREA: PI&R
January 8, 2002

EXHIBIT 6



January 8, 2002

EXHIBIT 7

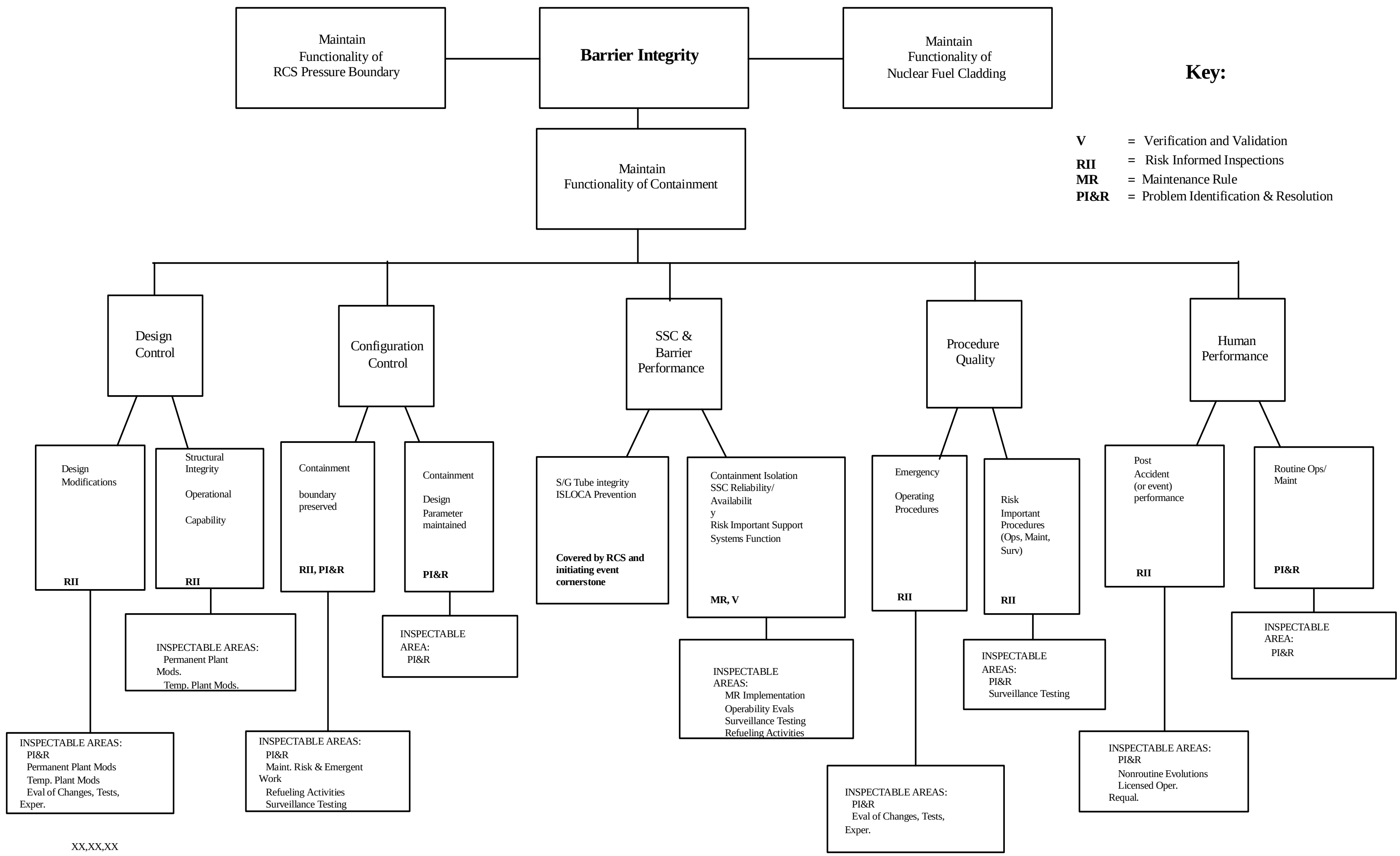

Maintain Functionality of RCS Pressure Boundary
Barrier Integrity
Maintain Functionality of Nuclear Fuel Cladding
Key:
V = Verification and Validation
RII = Risk Informed Inspections
MR = Maintenance Rule
PI&R = Problem Identification & Resolution
Maintain Functionality of Containment
Design Control
Configuration Control
SSC & Barrier Performance
Procedure Quality
Human Performance
Design Modifications
RII
Structural Integrity
Operational
Capability
RII
Containment
boundary preserved
RII, PI&R
Containment
Design Parameter maintained
PI&R
S/G Tube integrity
ISLOCA Prevention
Covered by RCS and initiating event cornerstone
Containment Isolation
SSC Reliability/ Availabilit y
Risk Important Support Systems Function
MR, V
Emergency
Operating Procedures
RII
Risk Important Procedures (Ops, Maint, Surv)
RII
Post Accident (or event) performance
RII
Routine Ops/ Maint
PI&R
INSPECTABLE AREAS:
Permanent Plant Mods.
Temp. Plant Mods.
INSPECTABLE AREA:
PI&R
INSPECTABLE AREAS:
MR Implementation
Operability Evals
Surveillance Testing
Refueling Activities
INSPECTABLE AREAS:
PI&R
Surveillance Testing
INSPECTABLE AREA:
PI&R
INSPECTABLE AREAS:
PI&R
Permanent Plant Mods
Temp. Plant Mods
Eval of Changes, Tests, Exper.
INSPECTABLE AREAS:
PI&R
Maint. Risk & Emergent Work
Refueling Activities
Surveillance Testing
INSPECTABLE AREAS:
PI&R
Eval of Changes, Tests, Exper.
INSPECTABLE AREAS:
PI&R
Nonroutine Evolutions
Licensed Oper. Requal.
XX,XX,XX

EXHIBIT 8

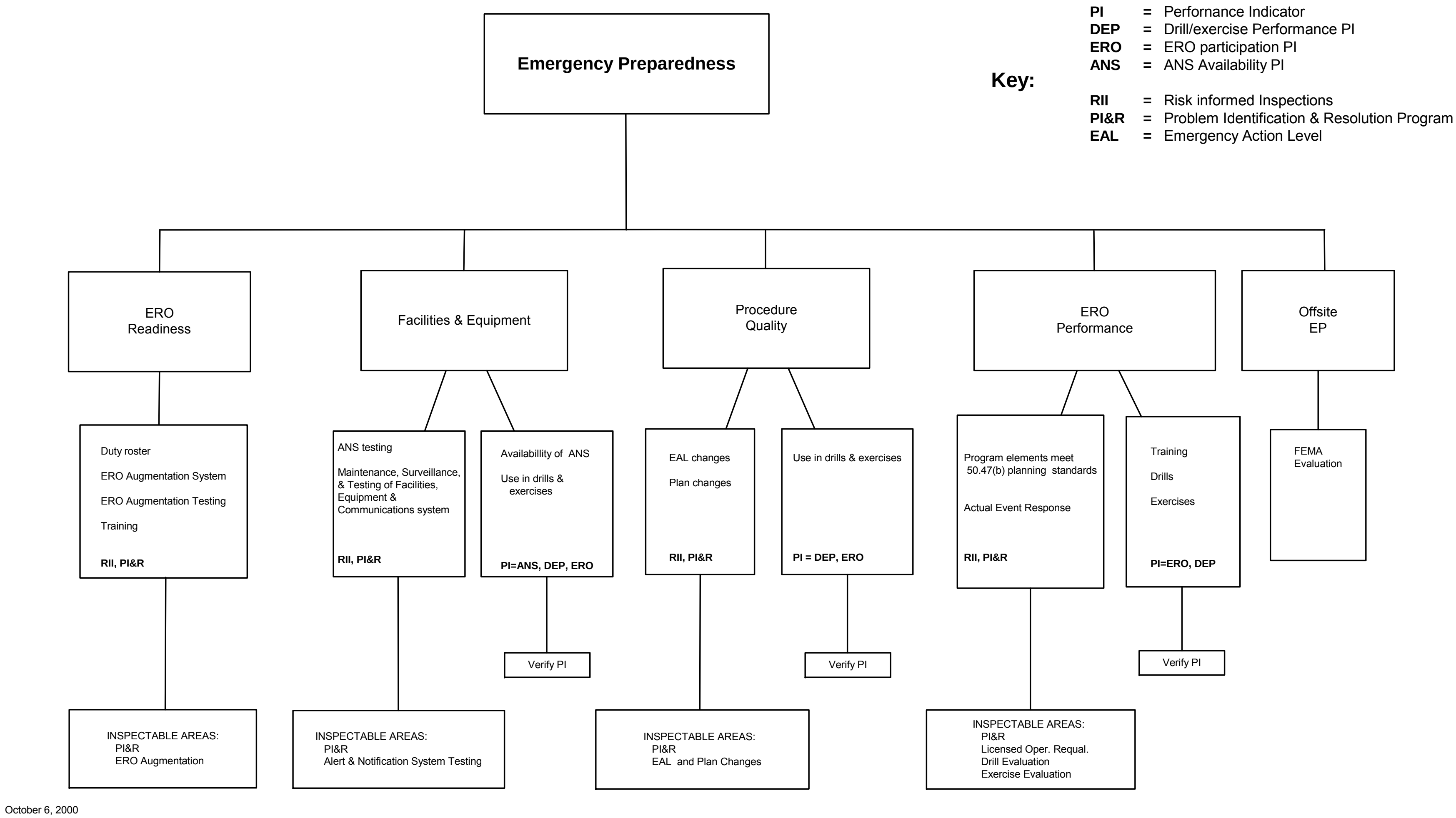
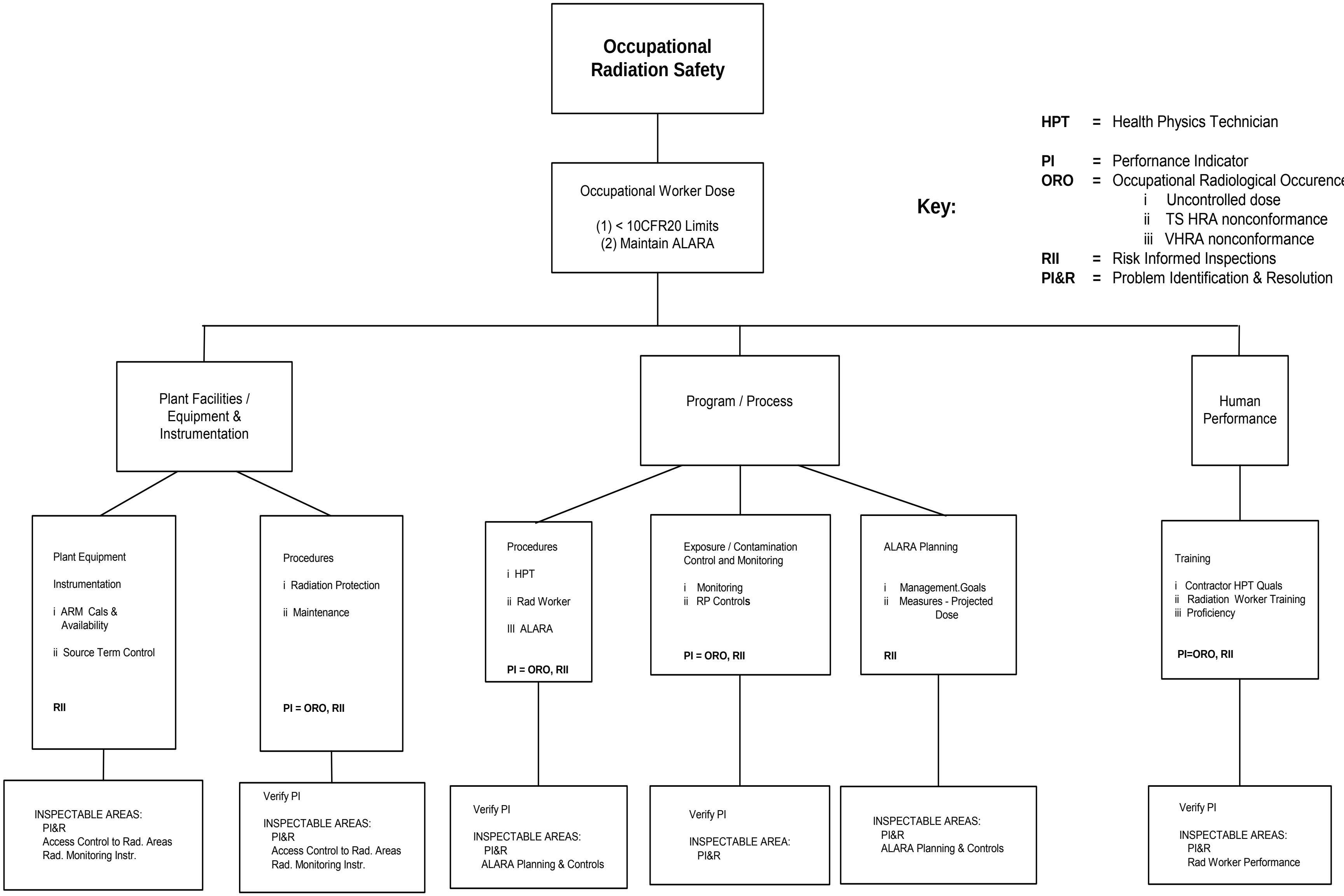

Key:
PI = Performance Indicator
DEP = Drill/exercise Performance PI
ERO = ERO participation PI
ANS = ANS Availability PI
RII = Risk informed Inspections
PI&R = Problem Identification & Resolution Program
EAL = Emergency Action Level
Emergency Preparedness
ERO Readiness
Facilities & Equipment
Procedure Quality
ERO Performance
Offsite EP
Duty roster
ERO Augmentation System
ERO Augmentation Testing
Training
RII, PI&R
ANS testing
Maintenance, Surveillance, & Testing of Facilities, Equipment & Communications system
RII, PI&R
Availabillity of ANS
Use in drills & exercises
PI=ANS, DEP, ERO
EAL changes
Plan changes
RII, PI&R
Use in drills & exercises
PI = DEP, ERO
Program elements meet 50.47(b) planning standards
Actual Event Response
RII, PI&R
Training
Drills
Exercises
PI=ERO, DEP
FEMA Evaluation
Verify PI
Verify PI
Verify PI
INSPECTABLE AREAS: PI&R ERO Augmentation
INSPECTABLE AREAS: PI&R Alert & Notification System Testing
INSPECTABLE AREAS: PI&R EAL and Plan Changes
INSPECTABLE AREAS: PI&R Licensed Oper. Requal. Drill Evaluation Exercise Evaluation
October 6, 2000

EXHIBIT 9



March 1, 2000

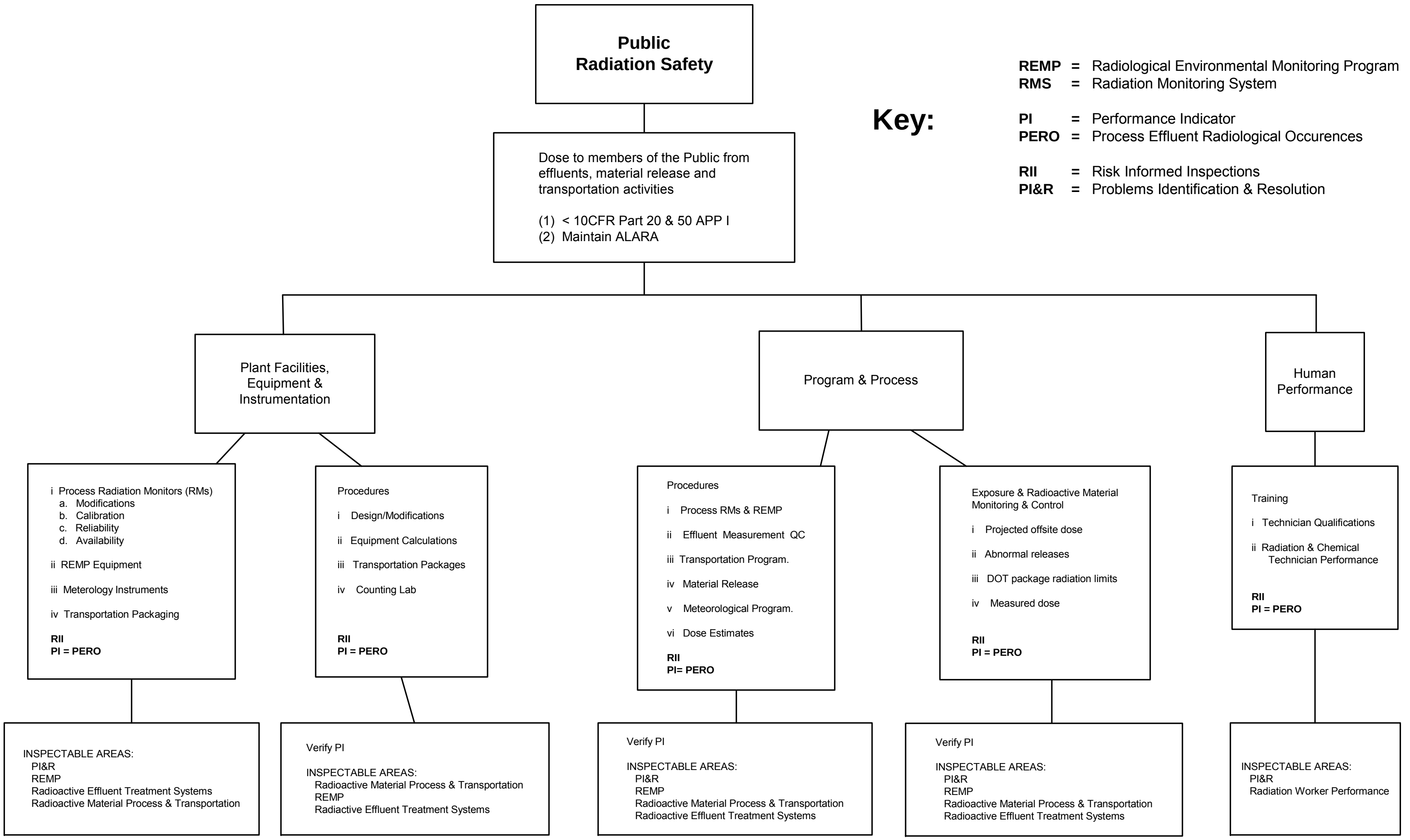
EXHIBIT 10

Key:

REMP = Radiological Environmental Monitoring Program
RMS = Radiation Monitoring System

PI = Performance Indicator
PERO = Process Effluent Radiological Occurences

RII = Risk Informed Inspections
PI&R = Problems Identification & Resolution

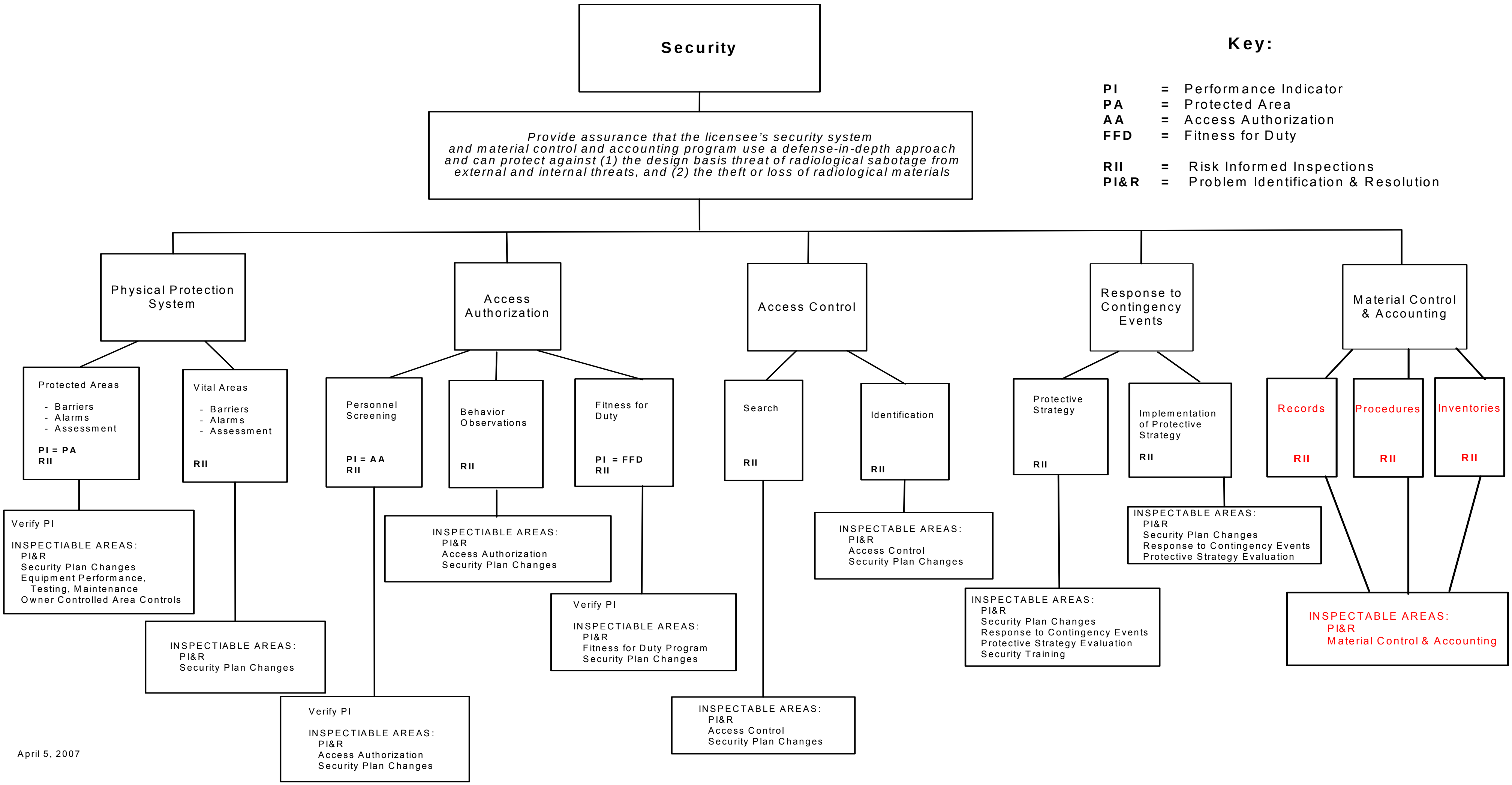


October 6, 2000

EXHIBIT 11

(The most recent version of this exhibit is located on-line in the NRC Reactor Oversight Process (ROP), Digital City, Forms, Templates, Sample Reports & More.)

Overview of the Security Cornerstone



April 5, 2007

EXHIBIT 12

CONCEPTUAL MODEL FOR EVALUATING LICENSEE PERFORMANCE

GREEN
Licensee Response Band

Cornerstone objectives fully met. Nominal risk with nominal deviation from expected performance

WHITE
Increased Regulatory Response Band

Cornerstone objectives met with *minimal* reduction in safety margin. Changes in performance consistent with $\Delta CDF<10^{-5}$ ($\Delta LERF<10^{-6}$).

YELLOW
Required Regulatory Response Band

Cornerstone objectives met with *significant* reduction in safety margin. Changes in performance consistent with $\Delta CDF<10^{-4}$ ($\Delta LERF<10^{-5}$)

RED
Significant Regulatory Response Band

Plant performance represents an unacceptable loss of safety margin. It should be noted that should licensee's performance result in a PI reaching the Red Band, margin would still exist before an undue risk to public health and safety would be presented.

Revision History For IMC 0308

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	10/16/06 CN 06-027	This IMC has been revised to incorporate comments from the Commission in which the term public confidence has been change to openness	None	N/A	N/A
N/A	11/08/07 CN 07-035	This IMC has been revised to incorporate changes in response to Feedback Forms 0308-0950, use of terms SCWE and safety culture, 0308-0952, remove containment PI from Exhibit 7, clarify definitions to performance band colors, and to revise reference numbering and remove/ move references to other portions of IMC 0308.	None	N/A	ML072830090