General Electric Advanced Technology Manual

Chapter 4.11

Regulatory Oversight Process

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

4.11 REGULATORY OVERSIGHT PROCESS

Learning Objectives

1. Recognize the purpose of the NRCs Regulatory Oversight Process.

2. Identify the three strategic performance areas and the cornerstones for each of the performance areas.

3. Identify the two methods that are used to measure plant performance and the associated performance bands assigned based on risk.

4. Given a copy of the NRC's Action Matrix and a plant's inspection findings and performance indicators, determine the appropriate regulatory response.

5. Identify the three Cross-Cutting Areas and criteria for establishing a substantive cross-cutting issue.

6. Recognize the differences between Baseline, Supplemental and Reactive inspections.

4.11.1 Purpose

The purpose of the Reactor Oversight Process is to objectively evaluate the overall performance of commercial nuclear power reactors to determine the appropriate level of agency response and to communicate the results to licensee management, members of the public and other government agencies.

The mission of the NRC is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment.

To help accomplish this mission, the NRC established strategic goals:

Safety: Ensure adequate protection of public health and safety and the environment.

Security: Ensure adequate protection in the secure use and management of radioactive materials.

The NRC has limited resources to accomplish these goals and established the Reactor Oversight Process (ROP) to both apply its resources where most appropriate and avoid placing an undue burden on the licensees.

To meet these objectives, the oversight process calls for:

- Focusing inspections on activities where the potential risks are greater.
- Applying greater regulatory attention to facilities with performance problems and reducing regulatory attention on plants that perform well.
- Using objective measures of the performance of plants whenever possible.
- Avoiding unnecessary regulatory burdens on nuclear facilities.
- Responding to violations of regulations in a predictable and consistent manner that reflects the safety impact of the violations.

4.11.2 Regulatory framework

The foundation of the ROP is the Regulatory Framework. The regulatory oversight framework developed by the staff is shown in Figure 4.11-1.

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 identified those aspects of licensee performance that are important to the mission and therefore merit regulatory oversight.

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.

The staff also identified safety culture aspects which were seen as "cross-cutting areas" which could potentially impact more than one cornerstone.

4.11.2.1 Reactor Safety Cornerstones

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 if multiple barriers are breached, a reactor accident could result which might compromise 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 (scrams) 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 monitor the availability, reliability, and capability of systems that mitigate the effects of initiating events to prevent core damage. Licensees reduce the likelihood of reactor accidents by maintaining the availability and reliability of mitigating systems. Mitigating systems include those systems associated with safety injection, decay 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 provide reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents. Licensees can reduce the effects of reactor accidents 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 licensees are capable of implementing adequate measures to protect public health and safety during a radiological emergency. Licensees provide reasonable assurance that their emergency preparedness program is effective through drills and exercises, participation in actual events, and testing of the Alert and Notification System (ANS). This cornerstone does not include the off-site actions, which are covered by FEMA.

4.11.2.2 Radiation Safety Cornerstones

Occupational Radiation Safety (Plant Worker)

The objective of this cornerstone is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine 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 As Low As Reasonably Achievable (ALARA) guidelines.

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 ALARA guidelines.

4.11.2.3 Safeguards Cornerstone

Security (Physical Protection)

The objective of this cornerstone is to provide assurance that the safeguards program will function to protect against the design basis threat of radiological sabotage. The threat could come from either external or internal sources. Licensees can maintain adequate protection against threats through an effective security program that relies on a defense in depth approach.

4.11.2.4 Cross-cutting Areas and Safety Culture

In addition to identifying the seven cornerstones of safety, the staff also identified certain areas of licensee performance that were seen as "cross-cutting" and potentially impacting more than one cornerstone. Performance in these areas is a measure of the licensee’s safety culture. The NRC’s definition of Safety Culture “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.” 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 weaknesses in these areas generally manifest themselves as the root causes of performance problems. Each of the three cross-cutting areas is comprised of different components, which are then further categorized into specific aspects of that component. Inspection Manual Chapter (IMC) 0310, Components Within The Cross-Cutting Areas, describes each area and its associated components and aspects in detail. The following sections provide an overview of each.

4.11.2.4.1 Human Performance Area (H)

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. 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. This area is divided into the following components and associated aspects:

1. **Decision-Making**. - Licensee decisions demonstrate that nuclear safety is an overriding priority.

    a. The licensee makes safety-significant or risk-significant decisions using a systematic process to ensure safety is maintained. H.1(a)

    b. The licensee uses conservative assumptions in decision making. H.1(b)

    c. The licensee communicates decisions and the basis for decisions to personnel. H.1(c)

2. **Resources** - The licensee ensures that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety.

    a. Long term plant safety through proper maintenance practices. H.2(a)

    b. Training of personnel and sufficient qualified personnel to maintain work hours within working hour guidelines. H.2(b)

    c. Complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. H.2(c)

d. Adequate and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities and equipment. H.2(d)

3. **Work Control** - The licensee plans and coordinates work activities, consistent with nuclear safety.

   a. The licensee appropriately plans work activities. H.3(a)

   b. The licensee appropriately coordinates work activities. H.3(b)

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

   a. The licensee communicates human error prevention techniques. H.4(a)

   b. The licensee defines and effectively communicates expectations regarding procedural compliance and personnel follow procedures. H.4(b)

   c. The licensee ensures supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. H.4(c)

4.11.2.4.2 Problem Identification and Resolution Area (P)

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. 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. This area is divided into the following components and associated aspects:

1. **Corrective Action Program** - 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):

   a. 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. P.1(a)

b. The licensee periodically trends and assesses information from the CAP and other assessments in the aggregate to identify programmatic and common cause problems. P.1(b)

c. The licensee thoroughly evaluates problems such that the resolutions address causes and extent of conditions, as necessary. P.1(c)

d. The licensee takes appropriate corrective actions to address safety issues and adverse trends in a timely manner. P.1(d)

e. If an alternative process for raising safety concerns exists, then it results in appropriate and timely resolutions of identified problems. P.1(e)

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

   a. The licensee systematically collects, evaluates, and communicates to affected internal stakeholders in a timely manner relevant internal and external OE. P.2(a)

   b. The licensee implements and institutionalizes OE through changes to station processes, procedures, equipment, and training programs. P.2(b)

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

   a. The licensee conducts self-assessments of sufficient depth that are comprehensive, objective, and self-critical. P.3(a)

   b. The licensee tracks and trends safety indicators which provide an accurate representation of performance. P.3(b)

   c. The licensee coordinates and communicates results from assessments to affected personnel, and takes corrective actions to address issues commensurate with their significance. P.3(c)

4.11.2.4.3 Safety Conscious Work Environment (S)

A Safety Conscious Work Environment (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. 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. This area is divided into the following components and associated aspects:

1. **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:

   a. 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, without fear of retaliation. S.1(a)

   b. If alternative processes for raising safety concerns or resolving differing professional opinions exist, then they are communicated, accessible, have an option to raise issues in confidence, and are independent, in the sense that the program does not report to line. S.1(b)

2. **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:

   a. 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. S.2(a)

   b. 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. S.2(b)

c. The potential chilling effects of disciplinary actions and other potentially adverse personnel actions are considered and compensatory actions are taken when appropriate. S.2(c)

4.11.3 Measuring Nuclear Plant Performance

Nuclear plant performance is measured by a combination of Performance Indicators (PIs) <u>and</u> findings resulting from the inspection program which will be focused on those plant activities which have the greatest impact on safety and overall risk.

4.11.3.1 Performance Indicators

Performance indicators and their associated thresholds were developed by the industry (NEI 99-02) and approved by the NRC (RIS 2000-08) when the revised Regulatory Oversight Process was developed. The performance indicator data is voluntarily submitted by the utilities each quarter. Even though submittal of data is voluntary, the NRC inspects the submittals for accuracy and can issue findings for inaccurate data if it impacted the NRCs ability to regulate the licensee.

Each of the performance indicators has criteria for measuring acceptable performance using a color-coded system for safety performance. Like all industrial activities, nuclear power plants are not error-free or risk-free. Equipment problems will occur. Each performance indicator is designed to determine acceptable levels of operation within adequate safety margins. **Green** indicates performance within an expected performance level where the associated cornerstone objectives are met. **White** represents performance outside an expected range of nominal utility performance but related cornerstone objectives are still being met. **Yellow** indicates related cornerstone objectives are being met, but with a moderate degradation in the safety margin. **Red** signals a significant reduction in safety margin in the area measured by the performance indicator. A summary of the indicators and their thresholds are provided in Table 4.11-1.

The indicators are compiled by the licensee and reported to the NRC on a quarterly basis. Following compilation and review by the NRC staff, the quarterly performance indicators are posted on the NRC's web site, with the exception of the Security Cornerstone PI. Figures 4.11-2 and 4.11-3 provide examples of the data posted. A summary of the PIs follows:

4.11.3.1.1 Initiating Events Performance Indicators

- **Unplanned Reactor Scrams** - The number of unplanned scrams during the previous four quarters, both manual and automatic, while critical per 7,000 hours. The scram rate is calculated per 7,000 critical hours because that value is representative of the critical hours of operation in a year for a typical plant.

- **Unplanned Power Changes** - The number of unplanned changes in reactor power of greater than 20% full-power, per 7,000 hours of critical operation excluding manual and automatic scrams.

- **Unplanned Scrams with Complications** – The total number of unplanned scrams while critical in the previous 4 quarters that required additional operator response. Such events or conditions have the potential to present additional challenges to the plant operations staff and therefore, may be more risk-significant than uncomplicated scrams. If the answer to any of the following questions is Yes, the unplanned scram counts as complicated as well as an unplanned scram.

  - Did an RPS actuation fail to indicate / establish a shutdown rod pattern for a cold clean core?
  - Was pressure control unable to be established following the initial transient?
  - Was power lost to any Class 1E Emergency / ESF bus?
  - Was a Level 1 Injection signal received?
  - Was Main Feedwater not available or not recoverable using approved plant procedures?
  - Following initial transient, did stabilization of reactor pressure/level and drywell pressure meet the entry conditions for EOPs?

4.11.3.1.2 Mitigating Systems Performance Indicators

- **Safety System Functional Failures** - The number of events or conditions that alone prevented, or could have prevented, the fulfillment of the safety function of structures or systems in the previous four quarters. By definition, these events are reportable under 10CFR50.73 as a loss of safety function.

- **Mitigating System Performance Indices (MSPIs)** - the sum of changes in a simplified core damage frequency evaluation resulting from differences in unavailability and unreliability of specific systems, relative to industry standard baseline values. The index calculates the change in core damage frequency (CDF) for the actual unreliability and unavailability of these systems compared with what was assumed in the licensee’s PRA. Each licensee maintains a MSPI Bases Document that documents the baseline values and assumptions used in calculating this PI. The systems monitored for these indicators in BWRs are:

- emergency AC power system
- high pressure injection system (high pressure coolant injection, high pressure core spray, or feedwater coolant injection)
- reactor core isolation cooling (or isolation condenser)
- residual heat removal system (suppression pool cooling)
- cooling water support system (includes direct cooling functions provided by service water and component cooling water or their cooling water equivalents for the above four monitored systems)

4.11.3.1.3 Barrier Integrity Performance Indicators

- **Reactor Coolant System (RCS) Activity** - The maximum monthly RCS activity in micro-Curies per gram (μCi/gm) dose equivalent Iodine-131 per the technical specifications, expressed as a percentage of the technical specification limit.

- **Reactor Coolant System (RCS) Leakage** - The maximum RCS Identified Leakage in gallons per minute each month as defined in Technical Specifications, expressed as a percentage of the technical specification limit. Note that for BWRs that do not have a Technical Specification limit on Identified Leakage, RCS Total Leakage is used.

4.11.3.1.4 Emergency Preparedness Performance Indicators

- **Drill/Exercise Performance** - The percentage of all drill, exercise, and actual opportunities that were performed timely and accurately during the previous eight quarters.

- **Emergency Response Organization (ERO) Drill Participation** - The percentage of key ERO members that have participated in a drill, exercise, or actual event during the previous eight quarters, as measured on the last calendar day of the quarter.

- **Alert and Notification System Reliability** - The percentage of ANS sirens that are capable of performing their function, as measured by periodic siren testing, in the previous 12 months. Periodic tests are the regularly scheduled tests that are conducted to actually test the ability of the sirens to perform their function (e.g., silent, growl, siren sound test).

4.11.3.1.5 Occupational Radiation Safety Performance Indicators

- **Occupational Exposure Control Effectiveness** - The performance indicator for this cornerstone is the sum of the following:
  - Technical specification high radiation area (>1 rem per hour) occurrences
  - Very high radiation area occurrences
  - Unintended exposure occurrences

4.11.3.1.6 Public Radiation Safety Performance Indicator

- **Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual (RETS/ODCM)** - Radiological effluent release occurrences per reactor unit that exceed the values listed below:

  Liquid Effluents
    - Whole Body - 1.5 mrem/qtr
    - Organ - 5 mrem/qtr

  Gaseous Effluents
    - Gamma Dose - 5 mrads/qtr
    - Beta Dose - 10 mrads/qtr
    - Organ Doses from I-131, I-133, H-3 & Particulates - 7.5 mrems/qtr

4.11.3.1.7 Physical Protection Performance Indicator

- **Protected Area (PA) Security Equipment** - PA Security equipment performance is measured by an index that compares the amount of time closed circuit television cameras (CCTVs) and intrusion detection system (IDS) are unavailable, as measured by compensatory hours, to the total hours in the period. A normalization factor is used to take into account site variability in the size and complexity of the systems.

4.11.3.2 Inspection Findings

In addition to Performance Indicators, the ROP uses the results from inspections to assess individual plant performance.

4.11.3.2.1 Types of Inspections

The inspection program is comprised of the following three major program elements:

1. Risk-Informed Baseline Inspection Program (Baseline) Inspections

2. Plant Specific Supplemental Inspections

3. Generic Safety Issue, Special, and Infrequent Inspections

4.11.3.2.1.1 Baseline Inspections

The inspection program is founded on “baseline” inspections common to all nuclear plants. The baseline inspection program, based on the cornerstones, focuses on activities and systems that are "risk significant," that is, those activities and systems that have a potential to trigger an accident, can mitigate the effects of an accident, or increase the consequences of a possible accident. The inspection areas were chosen because of their importance to potential risk, past operational experience, and regulatory requirements. The degree to which the area is measured by a performance indicator also affects the scope.

The baseline inspection program has three parts:

1. Inspection of areas not covered by performance indicators or where a performance indicator does not fully cover the inspection area;

2. Inspections to verify the accuracy of a licensee’s reports on performance indicators; and

3. A thorough review of the licensee’s effectiveness in finding and resolving problems on its own.

The inspections will be performed by NRC resident inspectors stationed at each nuclear power plant, and by regional-based and in some instances, headquarters-based inspectors. Table 4.11-2 provides a list of baseline inspections.

4.11.3.2.1.2 Supplemental Inspections

Inspections beyond the baseline program are set for plants with performance below established thresholds, as assessed through information gained from performance indicators and NRC inspections.

The supplemental element of the inspection program is designed to apply NRC inspection assets in an increasing manner when performance issues are identified, either by inspection findings evaluated using the significance determination process (SDP) or when performance indicator thresholds are exceeded. Depending on the risk significance and breadth of the identified performance issues, the supplemental inspections provide a graded response, which includes: oversight of the licensee’s root cause evaluation of the issues; expansion of the baseline inspection sample or a focused team inspection (as necessary to evaluate extent of condition); or a broad scope, multi-disciplined team inspection, which would include inspection of multiple cornerstone areas and inspection of crosscutting issues.

The supplemental inspection program contains three procedures which become deeper and broader as the safety significance of the performance issues increases. The criteria used for determining when these inspections are required, will be more fully discussed in later sections of this chapter on the assessment process and regulatory response.

For one or two white inputs in a strategic performance area (different cornerstones), supplemental inspection is limited to a thorough oversight of the licensee's evaluation. Inspection Procedure (IP) 95001 consists of review of the licensee's evaluation of root cause and extent of condition; plus review of proposed corrective actions. The inspection is limited to the specific issue(s) or performance area of concern.

For one degraded cornerstone (two white inputs or one yellow input) or any three white inputs in a strategic performance area, IP95002 will be performed. It consists of review of the licensee's evaluation of root cause and extent of condition, plus review of proposed corrective actions for both individual and collective issues. It also requires an independent NRC inspection to assess validity of licensee's extent of condition.

For repetitive degraded cornerstone, multiple degraded cornerstones, multiple yellow inputs, or any one red input, IP 95003 will be conducted. It consists of a large multi-disciplined NRC team inspection, focused on all key attributes associated with effected strategic performance areas. The intent of this procedure is to provide the NRC with supplemental information regarding licensee performance, as necessary to determine the breadth and depth of safety, organizational, and programmatic issues. As such, this procedure is more diagnostic than indicative, and includes reviews of programs and processes not inspected as part of the baseline inspection program. It also includes a review of a third-party independent assessment of the licensee's safety culture.

4.11.3.2.1.3 Generic Safety Issue, Special, and Infrequent Inspections

This category of inspection includes inspections of specific safety issues, inspections of major evolutions, milestones or unique capabilities; and inspections as a result of plant events or conditions.

Concerns with specific safety issues that arise may be addressed solely through the NRR license review process and the use of regulatory communications issued to licensees. If the concern is of safety significance, it may be appropriate to perform a one-time inspection under the safety issues program element. These inspections will be established by temporary instructions (TIs). For example, when it is determined that a safety issue addressed in a bulletin or generic letter requires inspection verification or follow-up, requirements and guidance for the inspection will be developed and issued in a TI. Unless such a TI is issued, inspection follow-up is not required to verify completion of licensees' actions discussed in a bulletin or generic letter. Recent examples of this type of inspection include:

TI 2515/173	Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative
TI 2515/176	Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing
TI 2515/177	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)
TI 2515/180	Inspection of Procedures and Processes for Managing Fatigue

The need may arise for specific inspections to address major evolutions limited to a few licensees or due to special capabilities. Examples include utilities that upgrade to digital control systems, replace major components, or receive approval for power up-rate, license renewal or installation of an Independent Spent Fuel Storage Facility. The need for these inspections will also be assessed on a case-by-case basis and conducted under the guidance of an inspection procedure or temporary instruction. Examples of this type of inspection include:

IP 52003	Digital Instrumentation & Control Modification Inspection
IP 60854.1	Preoperational Testing of ISFSIs at Operating Plants
IP 71003	Post-Approval Site Inspection for License Renewal
IP 71004	Power Uprate
IP 71007	Reactor Vessel Head Replacement Inspection

Also included in this category of inspection are **reactive inspections** conducted in response to significant plant events. A significant operational event is any radiological, safeguards, or other safety-related operational event at an NRC-licensed facility that poses an actual or a potential hazard to public health and safety, property, or the environment. The decision regarding an “investigatory response” for a significant operational event is defined by its risk significance, complexity, and generic safety implications. Significant operational power reactor events are evaluated on the basis of both deterministic criteria and risk significance such as conditional core damage probability (CCDP), in order to define the level of investigatory response.

An event will meet the deterministic criteria if the answers to one or more of the following questions for the event are Yes:

- Involved operations that exceeded, or were not included in, the design bases of the facility
- Involved a major deficiency in design, construction, or operation having potential generic safety implications
- Led to a significant loss of integrity of the fuel, the primary coolant pressure boundary, or the primary containment boundary of a nuclear reactor
- Led to the loss of a safety function or multiple failures in systems used to mitigate an actual event
- Involved possible adverse generic implications
- Involved significant unexpected system interactions
- Involved repetitive failures or events involving safety-related equipment or deficiencies in operations
- Involved questions or concerns pertaining to licensee operational performance

If the event meets the deterministic criteria, the risk significance is then determined in terms of conditional core damage probability. The results are then applied to the following table to assist NRC management in determining the level of response.

<table>
<tr><th colspan="10">Estimated CCDP</th></tr>
<tr><th colspan="2">CCDP < 1E-6</th><th colspan="2">1E-6 –> 1E-5</th><th colspan="2">1E-5 –> 1E-4</th><th colspan="2">1E-4 –> 1E-3</th><th colspan="2">CCDP > 1E-3</th></tr>
<tr><td colspan="3">No Additional Inspection</td><td colspan="7"></td></tr>
<tr><td colspan="2"></td><td colspan="4">SI</td><td colspan="4"></td></tr>
<tr><td colspan="4"></td><td colspan="5">AIT</td><td></td></tr>
<tr><td colspan="7"></td><td colspan="3">ITT</td></tr>
</table>

Incident Investigation Team (IIT) consists of technical experts who, to the extent practicable, do not have, and have not had, previous significant involvement with licensing and inspection activities at the affected facility and who perform the single NRC investigation of a significant operational event. An NRC senior manager leads the IIT. Each IIT reports directly to the Executive Director for Operations (EDO) and is independent of regional and headquarters office management. Guidance on the conduct of these inspections is contained in NUREG-1303, Incident Investigation Manual.

Augmented Inspection Team (AIT) consists of technical experts from the region in which the incident took place, augmented by personnel from headquarters or other regions or by contractors. The group performs an inspection of a significant operating event. AIT members may have had prior involvement with licensing and inspection activities at the affected facility. The AIT reports directly to the appropriate regional administrator. Guidance on the conduct of these inspections is contained in IP 93800, Augmented Inspection Team Implementing Procedure.

Special Inspection (SI) is similar to an AIT inspection except that the group generally is smaller (the number of members is based on management's judgment) and is generally not augmented by personnel from headquarters or other regions or by contractors. The special inspection team (SIT) reports directly to the appropriate regional administrator. Guidance on the conduct of these inspections is contained in IP 93812, Special Inspection.

4.11.3.2.2 Processing of Findings

During these inspections, issues of concern are identified by the inspectors. They must then determine if the issue constituted a **performance deficiency** (PD). A performance deficiency is an issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee's ability to foresee and correct, and therefore should have been prevented. A performance deficiency can exist if a licensee fails to meet a self-imposed standard or a standard required by regulation, thus a performance deficiency may exist independently of whether a regulatory requirement was violated. Failure to meet a regulatory requirement is considered a **violation** whereas failure to meet a standard (that is not required by regulation) is considered a **finding**.

Once it is has been established that the issue was a PD, it must then be evaluated to determine whether it was of minor significance or more-than-minor significance. IMC 0612, Appendix E provides a series of examples on issues that qualify as minor and those that are more-than-minor. If it is not possible to resolve whether the PD is minor or more-than-minor using Appendix E examples, the PD is evaluated directly against the minor screening questions and the answers to *all* of the questions must be "no" to be considered minor.

a. Could the PD be reasonably viewed as a precursor to a significant event?

b. If left uncorrected would the PD have the potential to lead to a more significant safety concern?

c. Does the PD relate to a performance indicator (PI) that would have caused the PI to exceed a threshold?

d. Is the PD associated with one of the cornerstone attributes and did the PD adversely affect the associated cornerstone objective?

If the performance deficiency is determined to be minor, it will not be documented in the inspection report. If the PD is determined to be more than minor, the significance of the issue must then be determined.

The methods for determining the risk significance of inspection findings is called the **significance determination process** (SDP). The SDP will assign a color (green, white, yellow, or red) to the finding. The color of the finding is indicative of the risk associated with the findings as indicated on Figure 4.11-4. IMC 0609 provides guidance on conducting significance determinations for various cornerstones. Many of the findings use a deterministic process of assigning risk; including emergency preparedness, fire protection and operator requalification. Table 4.11-3 lists the specific appendices used for various issues. For most findings the initial significance determination is normally performed by the inspector using the Phase 1, "Initial Screening and Characterization" worksheet described in Attachment 4 to IMC 0609. A portion of the worksheet for initiating events, mitigating systems and barrier integrity cornerstones is provided in Table 4.11-3. If the screening result is Green, then this would represent a final determination and will be characterized as Green at the exit meeting and in the inspection report. If the determination result is potentially White, Yellow, or Red, (i.e., greater-than-green) it will receive additional reviews by the regional staff, including Senior Reactor Analysts, using a Phase 2 and/or Phase 3 evaluation, to determine the final risk category.

Consistent with current practice, inspection reports will be issued following all inspections. The reports are available to the public. Additionally, the reports, along with any findings, are available on the NRC web site. Figure 4.11-8 provides an example of an inspection finding available on NRC public website.

4.11.3.2.3 Traditional Enforcement

Some violations are not conductive to SDP when they involve:

a. <u>Potentially Willful Violations.</u> The determination of willfulness is a legal decision that can only be made by the Office of General Council (OGC) using facts developed during an investigation conducted by the Office of Investigation (OI).

b. Violations that Impacted the Regulatory Process. The NRC considers the safety implications of violations that may impact the NRC's ability to carry out its statutory mission. Violations may be significant because they may challenge the regulatory envelope upon which certain activities were licensed. These types of violations include failures such as:

   i. Failure to provide complete and accurate information,
   ii. Failure to receive prior NRC approval for changes in licensed activities,
   iii. Failure to notify NRC of changes in licensed activities,
   iv. Failure to perform 10 CFR 50.59 analyses,
   v. Reporting failure, etc.,

c. Violation that Contributed to Actual Safety Consequences? Examples may include:

   i. actual onsite or offsite releases of radiation,
   ii. onsite or offsite radiation exposures,
   iii. accidental criticalities,
   iv. core damage,
   v. loss of significant safety barriers,
   vi. loss of control of radioactive material, or
   vii. radiological emergencies.

These violations are evaluated using the NRC Enforcement manual and are assigned severity levels (Level I, II, III or IV) depending on the significance. These violations may have underlying ROP issues of concern which are evaluated separately from the Traditional Enforcement (TE) violations. Because the TE violation is separated from the underlying finding and is not assigned an ROP color, it does not influence ROP Assessment.

4.11.4 Plant Performance Assessment

This assessment program collects information from inspections and performance indicators in order to enable the agency to arrive at objective conclusions about the licensee's safety performance. Based on this assessment information, the NRC determines the appropriate level of agency response, including supplemental inspection and pertinent regulatory actions ranging from management meetings up to and including orders for plant shutdown. Figure 4.11-5 illustrates the assessment process.

In addition, the assessment process also evaluates whether a Substantive Cross-Cutting Issue exists and what action the NRC will take if it does.

4.11.4.1 Performance Reviews

The assessment process consists of a series of reviews which are described below.

4.11.4.1.1 Continuous Review

The resident inspectors and branch chiefs in each regional office continuously monitor the performance of their assigned plants using the results of the PIs and inspection findings. Between the normal quarterly assessments, the region may issue an assessment follow-up letter and address an issue in accordance with the Action Matrix if: (1) a safety-significant inspection finding is finalized, or (2) a PI will cross a performance threshold at the end of the quarter based on current inputs. The assessment follow-up letter may be combined with the final SDP determination letter.

4.11.4.1.2 Quarterly Review

Each region conducts a quarterly review for each plant using PI data submitted by licensees and inspection findings compiled over the previous 12 months. The region determines the appropriate Action Matrix column for each plant and communicates the results to headquarters. The output of the quarterly review is a quarterly assessment follow-up letter. If there is no column change since the last assessment letter, a quarterly assessment follow-up letter is not required.

4.11.4.1.3 Mid-Cycle and End-of-Cycle Reviews

Each regional office conducts a mid-cycle and an end-of-cycle review for each plant using the most recent quarterly PIs, inspection findings, and enforcement actions compiled over the previous 12 months. The mid-cycle review meeting is chaired by a Division-level manager. The end-of-cycle review meeting is chaired by the regional administrator or his/her designee. Participating in these meetings are regional staff including DRP and DRS representatives, as well as headquarters staff.

The output of the mid-cycle review is a mid-cycle letter. The output from the end-of-cycle review is the Annual Assessment Letter. These letters contain:

- A summary of safety-significant PIs and inspection findings for the most recent two quarters as well as discussion of previous action taken by the licensee and the agency relative to these issues.
- Any changes in Action Matrix column status since the end of the previous cycle assessment period shall be noted.
- A qualitative discussion of substantive cross-cutting issues (SCCIs), if applicable.
- A discussion of non-SDP enforcement actions having Severity Level III or greater significance, including the planned Agency response.

- A discussion of findings that are currently being evaluated by the SDP that may affect the inspection plan.
- A statement of any actions to be taken by the agency in response to safety-significant issues, as well as any actions taken by the licensee.
- An inspection plan consisting of approximately 15 months.

The End-of-Cycle Summary Meeting is conducted following the conclusion of the end-of-cycle review meetings to summarize the results of the end-of-cycle review with the Director, NRR (or another member of the NRR Executive Team). The End-of-Cycle Summary Meeting is an informational meeting whose purpose is for regional management to engage headquarters management to ensure awareness of:

- Plants to be discussed at the Agency Action Review Meeting (AARM),
- Plants with significant performance issues,
- Plants with open Action Matrix deviations,
- Plants with substantive cross-cutting issues, and
- Agency actions already taken in response to plant performance.

Agency Action Review Meeting (AARM) is conducted several weeks after issuance of the annual assessment letters. This meeting is attended by appropriate senior NRC managers and is chaired by the Executive Director for Operations (EDO) or designee.

This meeting is a collegial review by senior NRC managers of:

- The appropriateness of agency actions for plants with significant performance issues based on data compiled during the end-of-cycle review and those that have moved into the “Multiple/Repetitive Degraded Cornerstone” or the “Unacceptable Performance” Columns during the first quarter of the year in which the AARM is held,
- Trends in overall industry performance,
- The appropriateness of agency actions concerning fuel cycle facilities and other materials licensees with significant performance problems,
- The results of the ROP self-assessment, including a review of approved deviations from the Action Matrix.

4.11.4.2 Action Matrix and Regulatory Response

During the assessment process the NRC determines the appropriate column that each unit is in per the Action Matrix and develops a regulatory oversight plan. The five columns of the Action Matrix and the associated regulatory response are listed below. A copy of the Action Matrix is provided in Table 4.11-5.

Licensee Response Column

- All Performance Indicators and Inspection Findings Green.
- Cornerstone objectives fully met.
- Routine resident inspector and staff interaction.
- Normal baseline inspection program.
- Annual assessment public meeting led by senior resident inspector or branch chief.

Regulatory Response Column

- One or two inputs White (in different cornerstones) in a Strategic Performance Area.
- Cornerstone objectives fully met.
- Licensee conducts root cause and develops corrective actions to address White inputs.
- Normal baseline inspection program and supplemental inspection 95001.
- Annual assessment public meeting led by branch chief or division director.

Degraded Cornerstone Column

- One degraded cornerstone (two inputs White or one input Yellow) or three White inputs in a Strategic Performance Area.
- Cornerstone objectives met with moderate degradation in safety performance.
- Licensee conducts cumulative root cause and develops corrective actions
- Normal baseline inspection program and supplemental inspection 95002.
- Regional Administrator to hold public meeting with licensee management

Multiple/Repetitive Degraded Cornerstone Column

- Repetitive degraded cornerstone, multiple degraded cornerstones, multiple Yellow inputs, or one Red input.
- Cornerstone objectives met with long standing issues or significant degradation in safety performance.
- Licensee develops performance improvement plan with NRC oversight.
- Normal baseline inspection program and supplemental inspection 95003.
- EDO to hold public meeting with senior licensee management.
- NRC issues Confirmatory Action Letter, Order (2.202), Demand for Information (2.204) or request for Additional Information (50.54(f)).

Unacceptable Performance Column

- Overall unacceptable performance.
- Plant not permitted to operate.
- Unacceptable margin to nuclear safety.
- Order to modify, suspend, or revoke license.
- Commission meeting with senior licensee management.

4.11.4.2.1 Deviations from the Action Matrix

The NRC recognizes that there may be rare instances in which the regulatory actions dictated by the Action Matrix may not be appropriate. In these instances, the Agency may deviate from the Action Matrix to either increase or decrease Agency action.

The EDO shall approve all deviations from the Action Matrix and inform the Commission when deviations are approved and annually at the Commission meeting on the results of the AARM. Memoranda requesting deviations from the Action Matrix should be initiated by the applicable regional administrator to the EDO and should go through the Office Director of NRR for program office approval. Any deviations from the Action Matrix shall be documented in the subsequent mid-cycle or annual assessment letter. Two examples of Action Matrix deviations are listed below:

- On September 29, 2010, the EDO approved a deviation from the ROP Action Matrix to allow Browns Ferry, Units 1, 2, and 3, to remain in Column 3 (Degraded Cornerstone Column) of the ROP Action Matrix in the fourth quarter of 2010 to provide additional time for the NRC to perform an IP 95002 supplemental inspection and follow-up assessment. The deviation was necessary because of high site activity level and regional inspection workload when the licensee informed the NRC that it was ready for the supplemental inspection. An initial interpretation of IMC 0305, “Operating Reactor Assessment Program,” regarding the definition of a repetitive degraded cornerstone also contributed to the need for the deviation. IMC 0305 will be clarified to avoid further misinterpretation. The staff intends to close the deviation by the end of the fourth quarter of 2010 after it completes the IP 95002 supplemental inspection and determines the appropriate applicable Action Matrix column and response.

- On April 5, 2010, the EDO approved a deviation from the ROP Action Matrix to provide increased oversight of the Vermont Yankee Nuclear Power Station related to on-site groundwater contamination and the March 1, 2010, demand for information (DFI). The deviation was necessary for the NRC to conduct agency actions beyond those prescribed by the site’s Action Matrix characterization. The staff intends to close the deviation when (1) the NRC has concluded that Entergy has established and implemented effluent control and environmental monitoring procedures that provide reasonable assurance that the existing groundwater conditions will continue to be effectively monitored and assessed; that the procedures will detect new or changed conditions in a timely manner; and that the procedures are sufficient to monitor remediation efforts associated with the Vermont Yankee groundwater contamination plume, and (2) the NRC has concluded that Entergy has appropriately addressed the information requested in the DFI, and that sufficient information is available for the NRC to determine whether further regulatory action, if any, is warranted.

4.11.4.3 Substantive Cross-Cutting Issues

During the mid-cycle and end-of-cycle reviews, the NRC also evaluates whether a substantive cross-cutting issue (SCCI) exists at a nuclear site. Each finding can have a cross-cutting aspect (CCA) assigned which reflects the root cause of the finding. A substantive cross-cutting issue exists if a cross-cutting theme exists and the NRC staff has a concern with the licensee's scope of efforts or progress in addressing the cross-cutting theme. For the cross-cutting areas of Human Performance (HU) and/or Problem Identification and Resolution (PI&R) a theme is established if four or more of these findings were assigned the same CCA from the twelve-month assessment period. For safety conscious work environment (SCWE) a cross-cutting theme exists if at least one of the following three conditions exists from an 18-month period prior to the assessment:

- There is a finding with a documented CCA in the area of SCWE and the impact on SCWE was not isolated.

- Licensee has received a chilling effect letter.

- The licensee has received correspondence from the NRC which transmitted (1) an enforcement action with a Severity Level of I, II, or III that involved discrimination or (2) a confirmatory order which involved discrimination.

If the assessment process determines that a SCCI exists, the assessment letter should summarize the specific SCCI by describing: The findings and their common cross-cutting aspects used to identify the SCCI; the single SCCI and each individual cross-cutting theme of that SCCI; the safety significance of the cross-cutting issue; the agency's action in the baseline program to monitor the issue, specifically, indicating how the staff will follow-up on the SCCI; the agency's assessment of the licensee's ability to address the SCCI or the licensee's progress to correct the issue; and the criteria for clearing the cross-cutting issue.

When the NRC identifies an SCCI 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.

4.11.5 Information Available to the Public

Information on plant performance is updated each quarter on the NRC's web site where performance histories and inspection findings are also available. Full inspection reports are available on the web site, in the NRC's online document collection called "ADAMS," and from the NRC's Public Document Room. The performance indicators and the assessment of inspection findings are placed on the NRC web site using the color notation of their significance green, white, yellow, or red. The statistics and NRC inspection findings which underlie the color notation are also posted on the web site.

The NRC maintains a ROP website which contains each unit's assessment, including PIs, findings, action matrix column and SCCIs. It also contains a detailed description of the oversight process. Figures 4.11-2, 3, 6, 7 and 8 provide a sampling of information that is available on the public website.

In essence, the program enhances public confidence in the NRC's regulatory program by increasing the predictability, consistency, objectivity and transparency of the oversight process.

4.11.6 Summary

The purpose of the Reactor Oversight Process (ROP) is to objectively evaluate the overall performance of commercial nuclear power reactors to determine the appropriate level of agency response and to communicate the results to licensee management, members of the public and other government agencies.

The ROP establishes Strategic Performance Areas and underlying cornerstones. 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 ROP uses inputs from performance indicators and inspection findings to assess performance. These inputs are then applied to the Action Matrix to determine the appropriate regulatory response. The baseline inspection program is intended to monitor plant activities and develop indicators of plant performance. If performance declines, the inspection effort would increase. Plants which do not meet the "safety cornerstone" objectives will receive supplemental inspections, focusing on the identified areas of declining performance. In addition, the ROP monitors the cross-cutting areas of human performance, problem identification and resolution and safety conscious work environment since deficiencies can impact all the cornerstones of safety and security.

4.11.7 References

- IMC 0305, Operating Reactor Assessment Program
- IMC 0308, Reactor Oversight Process (ROP) Basis Document
- IMC 0310, Components Within The Cross-Cutting Areas
- IMC 0609, Significance Determination Process (SDP)
- IMC 0612, Power Reactor Inspection Reports
- IMC 2515, Light Water Reactor Inspection Program -- Operations Phase
- MD 8.3, NRC Incident Investigation Program
- NUREG 1649, Reactor Oversight Process
- NEI 99-02, Regulatory Assessment Performance Indicator Guideline

Table 4.11-1 PERFORMANCE INDICATORS

Cornerstone	Indicator			Thresholds (see Note 1 and Note 2)		
				Increased Regulatory Response Band	**Required Regulatory Response Band**	**Unacceptable Performance Band**
Initiating Events	**IE01**	Unplanned Scrams per 7000 Critical Hours (automatic and manual scrams during the previous four quarters)		>3.0	>6.0	>25.0
	IE02	Unplanned Power Changes per 7000 Critical Hours (over previous four quarters)		>6.0	N/A	N/A
	IE03	Unplanned Scrams with Complications (over the previous four quarters)		>1	N/A	N/A
Mitigating Systems	**MS05**	Safety System Functional Failures (over previous four quarters)	BWRs PWRs	>6 >5	N/A N/A	N/A N/A
	MS06	Mitigating System Performance Index (Emergency AC Power Systems)		>1.0E-06 OR PLE = YES	>1.0E-05	>1.0E-04
	MS07	Mitigating System Performance Index (High Pressure Injection Systems)		>1.0E-06 OR PLE = YES	>1.0E-05	>1.0E-04
	MS08	Mitigating System Performance Index (Heat Removal Systems)		>1.0E-06 OR PLE = YES	>1.0E-05	>1.0E-04
	MS09	Mitigating System Performance Index (Residual Heat Removal Systems)		>1.0E-06 OR PLE = YES	>1.0E-05	>1.0E-04
	MS10	Mitigating System Performance Index (Cooling Water Systems)		>1.0E-06 OR PLE = YES	>1.0E-05	>1.0E-04
Barriers Fuel Cladding	**BI01**	Reactor Coolant System (RCS) Specific Activity (maximum monthly values, percent of Tech. Spec limit)		>50.0%	>100.0%	N/A
Reactor Coolant System	**BI02**	RCS Identified Leak Rate (maximum monthly values, percent of Tech. Spec. limit)		>50.0%	>100.0%	N/A

Table 4.11-1 PERFORMANCE INDICATORS (continued)

Cornerstone	Indicator		Thresholds (see Note 1 and Note 2)		
			Increased Regulatory Response Band	**Required Regulatory Response Band**	**Unacceptable Performance Band**
Emergency Preparedness	**EP01**	Drill/Exercise Performance (over previous eight quarters)	<90.0%	<70.0%	N/A
	EP02	ERO Drill Participation (percentage of Key ERO personnel that have participated in a drill or exercise in the previous eight quarters)	<80.0%	<60.0%	N/A
	EP03	Alert and Notification System Reliability (percentage reliability during previous four quarters)	<94.0%	<90.0%	N/A
Occupational Radiation Safety	**OR01**	Occupational Exposure Control Effectiveness (occurrences during previous 4 quarters)	>2	>5	N/A
Public Radiation Safety	**PR01**	RETS/ODCM Radiological Effluent Occurrence (occurrences during previous four quarters)	>1	>3	N/A
Physical Protection	**PP01**	Protected Area Security Equipment Performance Index (over a four quarter period)	>50.0%	N/A	N/A

Note 1: Thresholds that are specific to a site or unit will be provided in Appendix D when identified.

Note 2: PLE – System Component Performance Limit Exceeded

Table 4.11-2 Baseline Inspection Procedures

IP No.	Title	Frequency[1]
71111 Reactor Safety – Initiating Events, Mitigating Systems, Barrier Integrity		
71111.01	Adverse Weather Protection	A
71111.04	Equipment Alignment	Q/A
71111.05AQ	Fire Protection	Q/A
71111.05T	Fire Protection	T
71111.05TTP	Fire Protection – NFPA 805 Transition Period (Triennial)	T
71111.06	Flood Protection Measures	A
71111.07	Heat Sink Performance	A/T
71111.08	In-service Inspection Activities	R
71111.11	Licensed Operator Requalification Program	Q/B
71111.12	Maintenance Effectiveness	A
71111.13	Maintenance Risk Assessment and Emergent Work Control	A
71111.15	Operability Evaluations	A
71111.17	Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications	T
71111.18	Plant Modifications	A
71111.19	Post –Maintenance Testing	A
71111.20	Refueling and Outage Activities	R
71111.21	Component Design Bases Inspection	T
71111.22	Surveillance Testing	A
71114 Reactor Safety – Emergency Preparedness		
71114.01	Exercise Evaluation	B
71114.02	Alert Notification System Testing	B
71114.03	Emergency Response Organization Staffing and augmentation System	B
71114.04	Emergency Action Level and Emergency Plan Changes	A
71114.05	Correction of Emergency Preparedness Weaknesses	B
71114.06	Drill Evaluation	A
71114.07	Force-On-Force (FOF) Exercise Evaluation	T
71121 Occupational Radiation Safety		
71121.01	Access Control to Radiologically Significant Areas	A
71121.02	ALARA Planning and Controls	B
71121.03	Radiation Monitoring	B

Table 4.11-2 Baseline Inspection Procedures (continued)

71122 Public Radiation Safety		
71122.01	Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems	B
71122.02	Radioactive Material Processing and Transportation	B
71122.03	Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program	B
71130 Security		
71130.01	Access Authorization	B
71130.02	Access Control	A
71130.03	Contingency Response – Force-On-Force Testing	T
71130.04	Equipment Performance, Testing and Maintenance	B
71130.05	Protective Strategy Evaluation	T
71130.07	Security Training	B
71130.08	Fitness-For-Duty Program	B
71130.09	Owner Controlled Area Controls	A
71130.10	Information Technology Security	TBD
71130.11	Materials Control and Accountability	TBD
71130.12	Physical Protection of Shipments of Irradiated Fuel	TBD
Other Baseline Procedures		
71150	Discrepant or Unreported Performance Indicator Data	AN
71151	Performance Indicator Verification	A
71152	Identification and Resolution of Problems	A/B
71153	Event Followup	AN

Q = Quarterly,
A = Annual,
R = Refueling Outage,
B = Biennial,
T = Triennial
AN = As Needed

Table 4.11-3 SDP Guidance in IMC 0609 Appendices

Appendix A	Significance Determination of Reactor Inspection Findings for At-Power Situations
Appendix B	Emergency Preparedness SDP
Appendix C	Occupational Radiation Safety SDP
Appendix D	Public Radiation Safety SDP
Appendix E	Part 1, Baseline Security SDP for Power Reactors and Part II, Force-on-Force Security SDP for Power Reactors
Appendix F	Fire Protection and Post-Fire Safe Shutdown SDP
Appendix G	Shutdown Safety SDP
Appendix H	Containment Integrity SDP
Appendix I	Operator Requalification, Human Performance
Appendix J	Steam Generator Tube Integrity SDP
Appendix K	Maintenance Risk Assessment and Risk Management SDP
Appendix M	Significance Determination Process Using Qualitative Attributes

Table 4.11-4 Phase 1 SDP WORKSHEET FOR IE, MS, and BI CORNERSTONES

Initiating Events Cornerstone	Mitigation Systems Cornerstone	RCS or Fuel Barrier	Containment Barrier
LOCA Initiators 1. Assuming worst case degradation, would the finding result in exceeding the Tech Spec limit for any RCS leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function. ☐ **If YES ➛Stop. Go to Appendix A.** ☐ If NO, screen as Green. Transient Initiators 1. Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available? ☐ **If YES➛Stop. Go to Appendix A.** ☐ If NO, screen as Green. External Event Initiators 1. Does the finding increase the likelihood of a fire or internal/external flood? ☐ **If YES** ➛ Use the IPEEE or other existing plant-specific analyses to identify core damage scenarios of concern and factors that increase the frequency. Provide this input for Phase **3** analysis. ☐ If NO, screen as Green.	1. Is the finding a design or qualification deficiency confirmed not to result in loss of operability or functionality? ☐ **If YES, screen as Green.** ☐ If NO, continue. 2. Does the finding represent a loss of system safety function? ☐ **If YES ➛Stop. Go to Appendix A.** ☐ If NO, continue. 3. Does the finding represent actual loss of safety function of a single Train, for > it's Tech Spec Allowed Outage Time? ☐ **If YES ➛ Stop. Go to Appendix A.** ☐ If NO, continue. 4. Does the finding represent an actual loss of safety function of one or more non-Tech Spec Trains of equipment designated as risk-significant per 10CFR50.65, for >24 hrs? ☐ **If YES ➛ Stop. Go to Appendix A.** ☐ If NO, continue. 5. Does the finding screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event, using the criteria on page 5 of this Worksheet? ☐ **If YES** ➛ Use the IPEEE or other existing plant-specific analyses to identify core damage scenarios of concern and provide this input for **Phase 3** analysis. ☐ If NO, screen as Green.	RCS Barrier (e.g., pressurized thermal shock issues) ☐ **Stop. Go to Phase 3**. Fuel Barrier ☐ Stop. Screen as Green. Spent Fuel Pool Issues 1. Does the finding result in loss of cooling to the spent fuel pool, whereby operator or equipment failures could preclude restoration of cooling prior to pool boiling? ☐ **If YES** ➛ Stop. Go to **Appendix M.** ☐ If NO, continue. 2. Does the finding result from fuel handling errors that caused damage to fuel clad integrity or a dropped assembly (includes ISFSI)? ☐ **If YES** ➛ Stop. Go to **Appendix M**. ☐ If NO, continue. 3. Does the finding result in a loss of spent fuel pool inventory greater than 10% of SFP volume? ☐ **If YES** ➛ Stop. Go to **Appendix A**. ☐ If NO, screen as Green.	1. Does the finding only represent a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, or SBGT system (BWR)? ☐ **If YES** ➛ screen as Green. ☐ If NO, continue. 2. Does the finding represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere? ☐ **If YES ➛ Stop. Go to Phase 3.** ☐ If NO, continue. 3. Does the finding represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, containment isolation system (logic and instrumentation), and heat removal components? ☐ **If YES ➛ Stop. Go to Appendix H.** ☐ If NO, continue. 4. Does the finding involve an actual reduction in function of hydrogen ignitors in the reactor containment? ☐ **If YES ➛ Stop. Go to Appendix H.** ☐ If NO, screen as Green.

Table 4.11-5 ACTION MATRIX

		Licensee Response Column	Regulatory Response Column	Degraded Cornerstone Column	Multiple/ Repetitive Degraded Cornerstone Column	Unacceptable Performance Column	IMC 0350 Process[1]
RESULTS		All Assessment Inputs (Performance Indicators (PIs) and Inspection Findings) Green; Cornerstone Objectives Fully Met	One or Two White Inputs (in different cornerstones) in a Strategic Performance Area; Cornerstone Objectives Fully Met	One Degraded Cornerstone (2 White Inputs or 1 Yellow Input) or any 3 White Inputs in a Strategic Performance Area; Cornerstone Objectives Met with Moderate Degradation in Safety Performance	Repetitive Degraded Cornerstone, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or 1 Red Input; Cornerstone Objectives Met with Longstanding Issues or Significant Degradation in Safety Performance	Overall Unacceptable Performance; Plants Not Permitted to Operate Within this Band, Unacceptable Margin to Safety	Plants in a shutdown condition with performance problems placed under the IMC 0350 process
RESPONSE	Regulatory Performance Meeting	None	Branch Chief (BC) or Division Director (DD) Meet with Licensee	Regional Administrator (RA) (or Designee) Meet with Senior Licensee Management.	EDO/DEDO (or Designee) meet with Senior Licensee Management	EDO/DEDO (or Designee) Meet with Senior Licensee Management	RA/EDO (or Designee) Meet with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee Root cause Evaluation and corrective action with NRC Oversight	Licensee cumulative root cause evaluation with NRC Oversight	Licensee Performance Improvement Plan with NRC Oversight		Licensee Performance Improvement Plan / Restart Plan with NRC Oversight
	NRC Inspection	Risk-Informed Baseline Inspection Program	Baseline and supplemental inspection procedure 95001	Baseline and supplemental inspection procedure 95002	Baseline and supplemental inspection procedure 95003		Baseline and Supplemental as Practicable, Plus Special Inspections per Restart Checklist.
	Regulatory Actions[2]	None	Supplemental inspection only	Supplemental inspection only Plant Discussed at AARM if Conditions Met	-10 CFR 2.204 DFI -10 CFR 50.54(f) Letter - CAL/Order Plant Discussed at AARM	Order to Modify, Suspend, or Revoke Licensed Activities Plant Discussed at AARM	CAL/Order Requiring NRC Approval for Restart. Plant Discussed at AARM
COMMUNICATION	Assessment Letters	BC or DD review/sign assessment report (w/ inspection plan)	DD review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)		N/A. RA (or 0350 Panel Chairman) Review/ Sign 0350-Related Correspondence
	Annual Involvement of Public Stakeholders	Various public stakeholder options (see section 09) involving the SRI or BC	Various public stakeholder options (see section 09) involving the BC or DD	RA (or Designee) Discuss Performance with Senior Licensee Management	EDO/DEDO (or Designee) Discuss Performance with Senior Licensee Management		N/A. 0350 Panel Chairman Conduct Public Status Meetings Periodically
	Commission Involvement	None	None	Possible Commission Meeting if Licensee Remains for 3 yrs	Commission Meeting with Senior Licensee Management Within 6 mo.	Commission Meeting with Senior Licensee Management	Commission Meetings as Requested, Restart Approval in Some Cases.
	SAFETY SIGNIFICANCE INCREASED ---------->						

[1] The IMC 0350 Process column is included for illustrative purposes only and is not necessarily representative of the worst level of licensee performance. Plants under the IMC 0350 oversight process are considered outside the auspices of the ROP Action Matrix. See IMC 0350, "Oversight of Reactor Facilities in a Shutdown Condition due to Significant Performance and/or Operational Concerns," for more detail.

[2] Other than the CAL, the regulatory actions for plants in the Multiple/Repetitive Degraded Cornerstone column and IMC 0350 column are not mandatory agency actions. However, the regional office should consider each of these regulatory actions when significant new information regarding licensee performance becomes available.

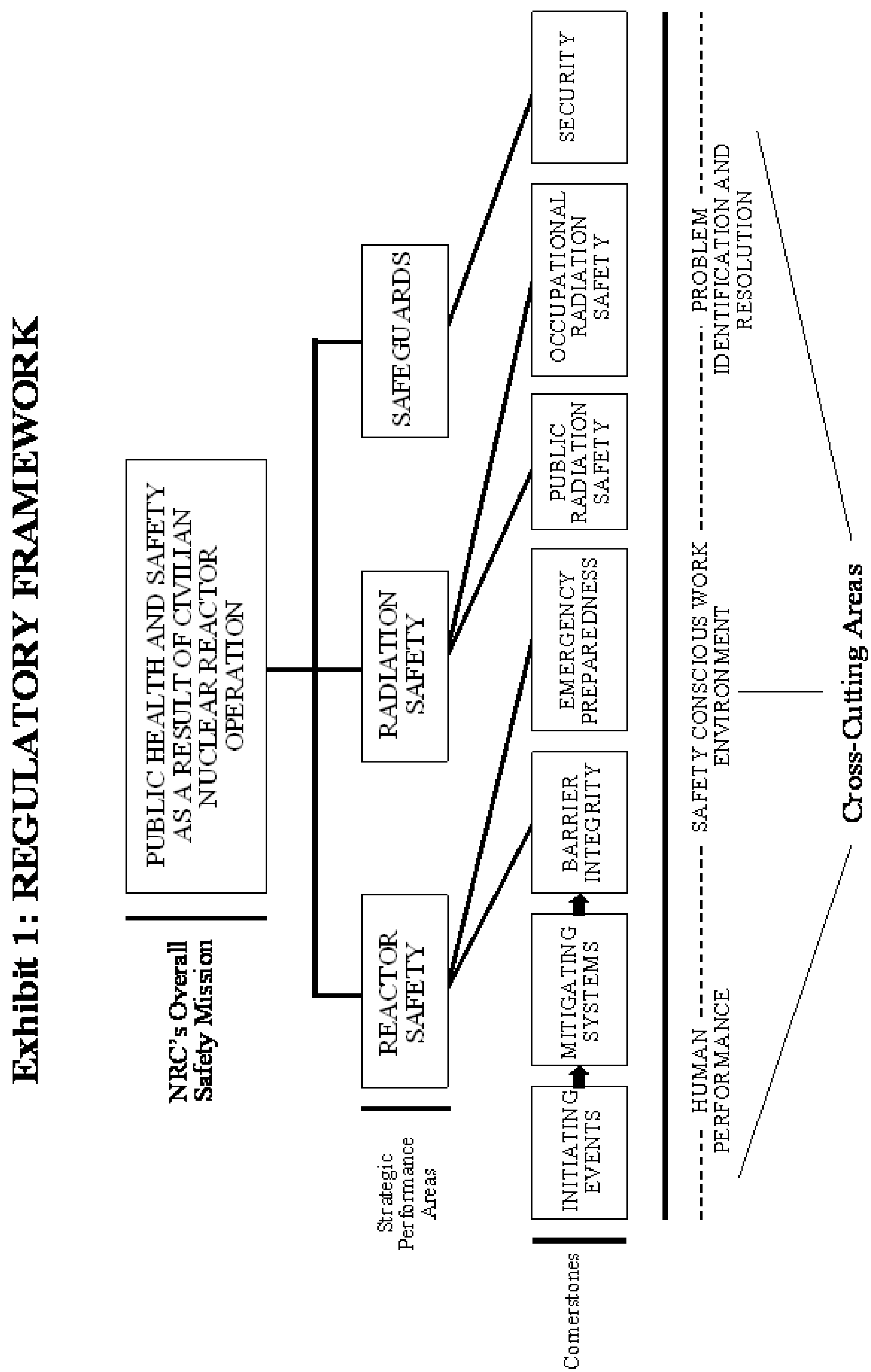


Figure 4.11-1 Regulatory Framework

Index | Site Map | FAQ | Facility Info | Reading Rm | New | Help | Glossary | Contact Us

BrowseAloud

Search

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

About NRC | Nuclear Reactors | Nuclear Materials | Radioactive Waste | Nuclear Security | Public Meetings & Involvement

Home > Nuclear Reactors > Operating Reactors > Oversight > Reactor Oversight Process

3Q/2010 ROP Performance Indicators Summary

Security information not publicly available.

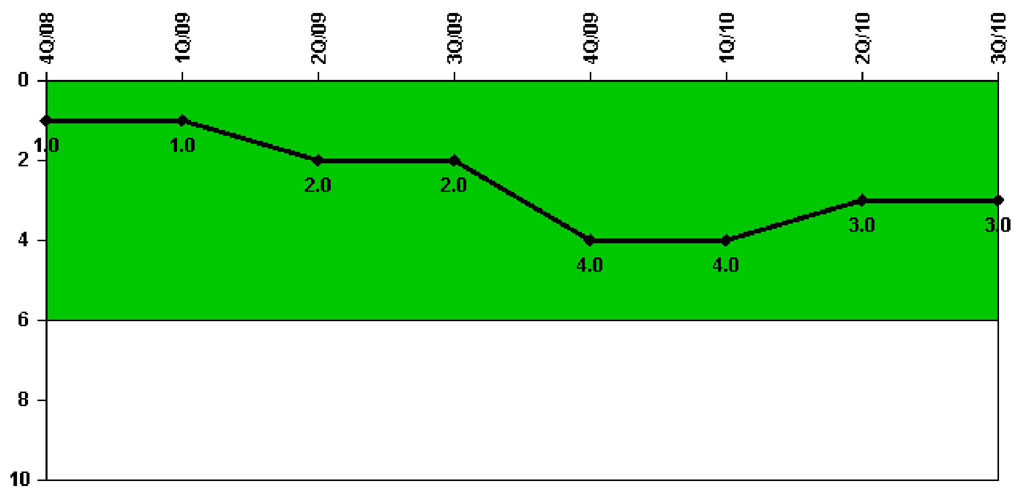
Plants	IE 01	IE 03	IE 04	MS 05	MS 06	MS 07	MS 08	MS 09	MS 10	BI 01	BI 02	EP 01	EP 02	EP 03	OR 01	PR 01
Arkansas Nuclear 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Arkansas Nuclear 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Beaver Valley 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Beaver Valley 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Figure 4.11-2 Performance Indicator Summary

Browns Ferry 1

3Q/2010 Performance Indicators

Safety System Functional Failures (BWR)



Thresholds: White > 6.0

Notes

Safety System Functional Failures (BWR)	4Q/08	1Q/09	2Q/09	3Q/09	4Q/09	1Q/10	2Q/10	3Q/10
Safety System Functional Failures	1	0	1	0	3	0	0	0
Indicator value	**1**	**1**	**2**	**2**	**4**	**4**	**3**	**3**

Licensee Comments: none

Figure 4.11-3 Individual Performance Indicator Results

- Green – Very Low Safety Significance.
- White – Low To Moderate Safety Significance.
- Yellow – Substantial Safety Significance.
- Red – High Safety Significance.

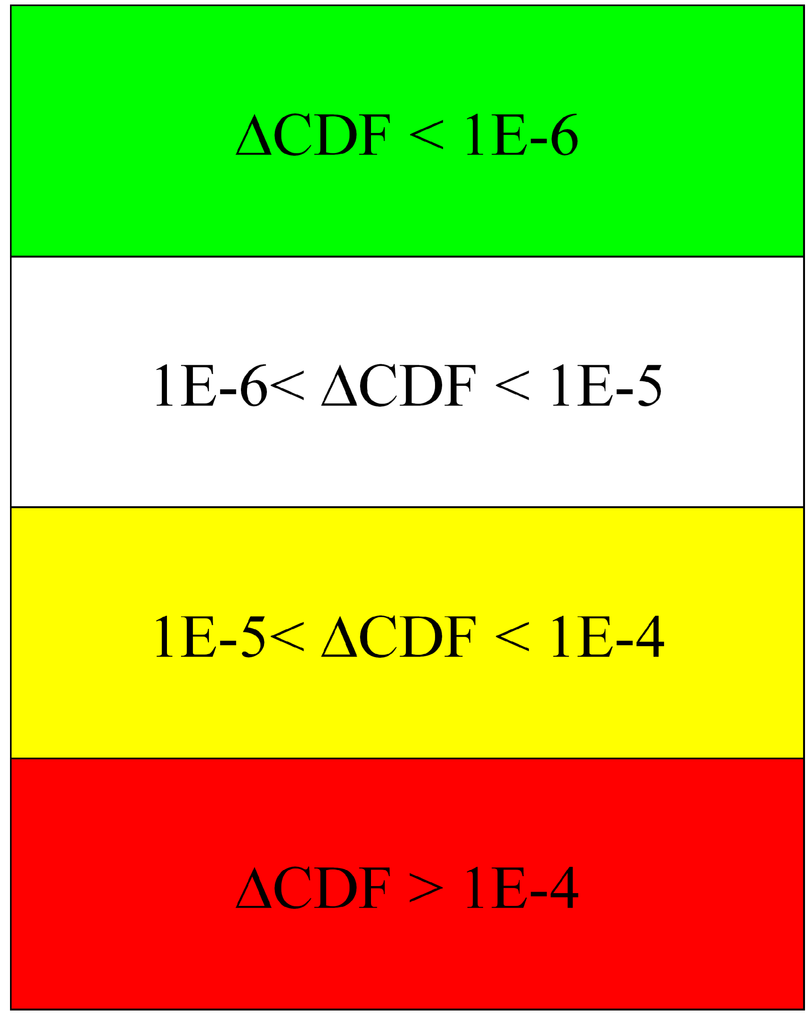


Figure 4.11-4 Performance Bands

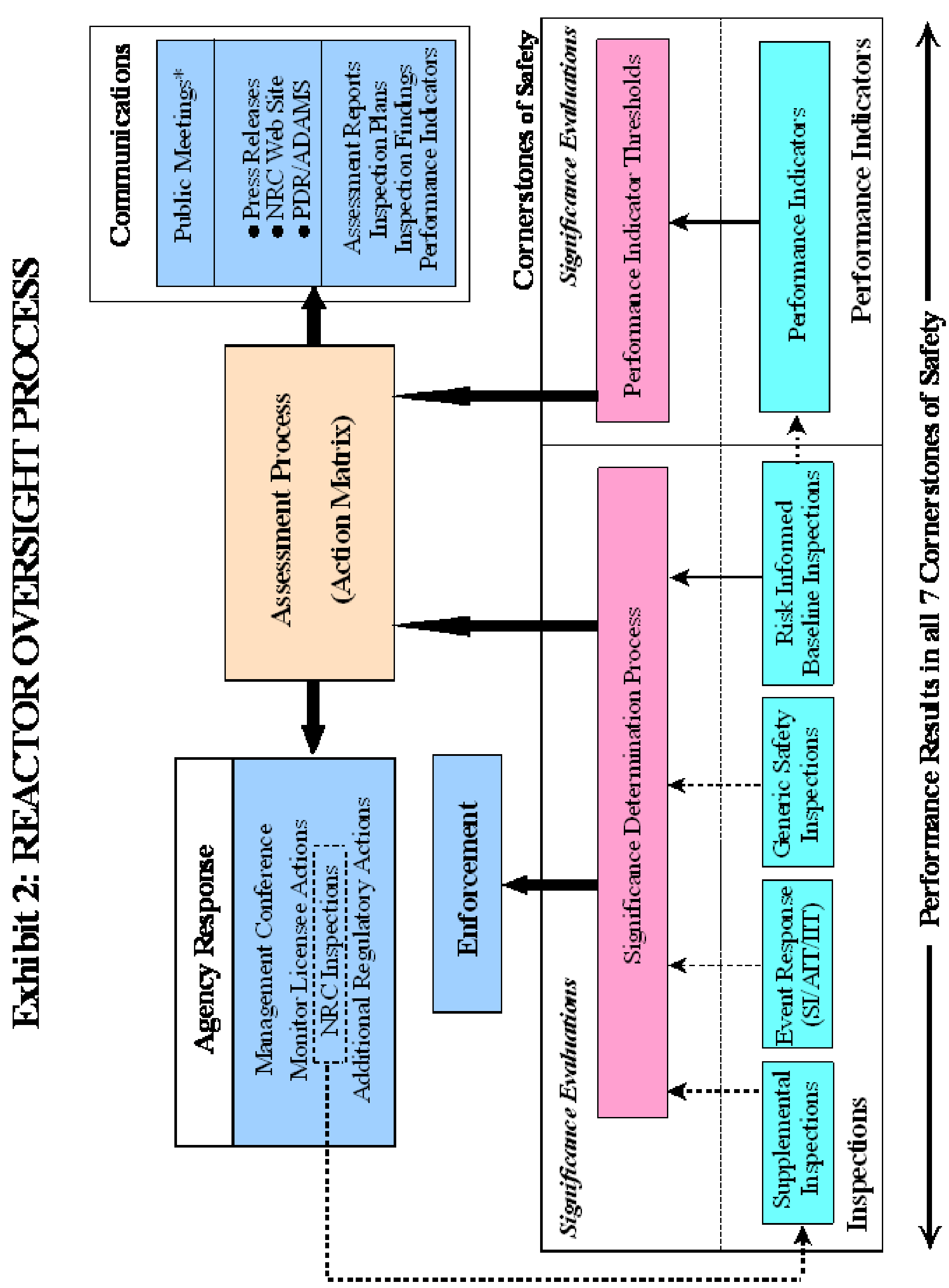


Figure 4.11-5 Reactor Oversight Process

Index | Site Map | FAQ | Facility Info | Reading Rm | New | Help | Glossary | Contact Us

BrowseAloud

Search

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

About NRC | Nuclear Reactors | Nuclear Materials | Radioactive Waste | Nuclear Security | Public Meetings & Involvement

Home > Nuclear Reactors > Operating Reactors > Oversight > Reactor Oversight Process

Mid Cycle 2010 ROP Substantive Cross Cutting Issues Summary

The NRC evaluates whether a substantive cross-cutting issue exists at each operating reactor twice a year; during the mid-cycle and end-of-cycle assessment meetings in accordance with IMC 0305. If the NRC determines that a substantive cross-cutting issue exists at a given plant, the resultant mid-cycle and end-of-cycle assessment letters summarize the specific substantive cross-cutting issue to include the necessary actions to resolve the issue. The next mid-cycle or annual assessment letter will either state that the issue has been satisfactorily resolved or summarize the agency's assessment and licensee's progress in addressing the issue.

"Yes" in the matrix below indicates that the plant has a open substantive cross-cutting issue in that area based on the most recent mid-cycle or end-of-cycle assessment. Clicking on the "Yes" will take you directly to the assessment letter that describes the issue. "Closed" in the matrix below indicates that the plant had a substantive cross-cutting issue closed based on the last assessment. Clicking on the "Closed" will take you directly to the assessment letter that describes the issue.

Plants	Human Performance	Problem Identification and Resolution	Safety Conscious Work Environment
Arkansas Nuclear 1			
Arkansas Nuclear 2			
Beaver Valley 1			
Beaver Valley 2			
Braidwood 1	Closed		
Braidwood 2	Closed		
Browns Ferry 1		Yes	
Browns Ferry 2		Yes	
Browns Ferry 3		Yes	
Brunswick 1			
Brunswick 2			
Byron 1			
Byron 2			
Callaway			
Calvert Cliffs 1			
Calvert Cliffs 2			
Catawba 1			
Catawba 2			
Clinton			
Columbia Generating Station			
Comanche Peak 1			
Comanche Peak 2			
Cooper	Yes		
Crystal River 3			
D.C. Cook 1			
D.C. Cook 2			
Davis-Besse			
Diablo Canyon 1		Yes	
Diablo Canyon 2		Yes	
Dresden 2	Closed		
Dresden 3	Closed		
Duane Arnold			
Farley 1	Yes		
Farley 2	Yes		

Figure 4.11-6 SCCI Summary

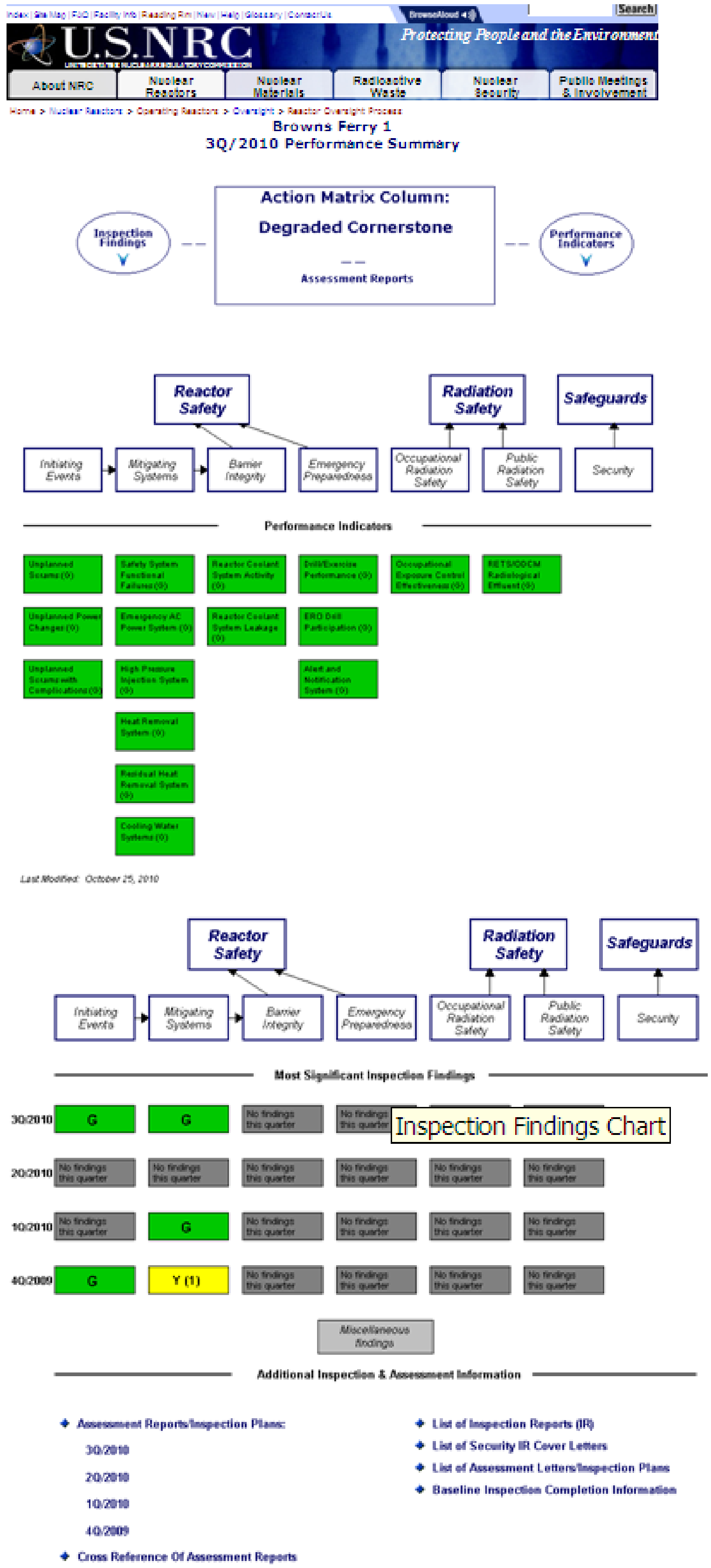


Figure 4.11-7 Individual Plant Performance Summary

Index | Site Map | FAQ | Facility Info | Reading Rm | New | Help | Glossary | Contact Us

BrowseAloud

Search

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

About NRC | Nuclear Reactors | Nuclear Materials | Radioactive Waste | Nuclear Security | Public Meetings & Involvement

Home > Nuclear Reactors > Operating Reactors > Oversight > Reactor Oversight Process

Browns Ferry 1
3Q/2010 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to adequately test molded case circuit breakers
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for failure to establish a preventive maintenance (PM) test program for safety-related molded case circuit breakers (MCCBs) to demonstrate these breakers would perform satisfactorily upon demand. Since initial startup of all three units, the inspectors found that the licensee had not included 612 critical MCCBs, many of them safety-related, in their PM program which resulted in the MCCBs receiving no planned maintenance or testing. The licensee entered this issue into the corrective action program as problem evaluation report (PER) 209095. The licensee's corrective actions included: identifying all critical MCCBs that required preventive maintenance, developing test procedures for these MCCBs, performing testing for all affected MCCBs, and conducting an extent-of-condition review of all safety-related components potentially excluded from the PM program.

This finding was determined to be of greater than minor significance because it was associated with the Protection Against External Factors attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events, such as fire, that challenge critical safety functions during shutdown as well as power operations. Specifically, the lack of a PM program for safety-related MCCBs resulted in no periodic planned maintenance or testing being performed since original installation, which in most cases was over thirty years. Based on operating experience, this could result in a breaker being slow to trip or sticking in the "on" position after an over-current condition. In accordance with IMC 0609, Significance Determination Process (SDP), Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to require a Phase 3 analysis since the finding represented an increase in the likelihood of a fire caused by an electrical fault at the MCCB compartment with the breaker not opening. A regional Senior Reactor Analyst conducted a Phase 3 SDP analysis, which concluded that the finding was of very low safety significance (Green).
The cause of this finding was directly related to the cross cutting aspect of Appropriate Corrective Actions in the Corrective Action Program component of the Problem Identification and Resolution area, because the licensee did not adequately implement corrective actions to resolve the deficiencies previously identified by PER 131875 regarding certain Westinghouse MCCBs that were not in the PM program [P.1(d)]. (Section 4OA5.4)

Inspection Report# : 2010004 *(pdf)*

Figure 4.11-8 Inspection Findings