Safety Culture in the ROP

Table of Contents

1. Objectives of this Resource

Upon completion of this training, participants should be able to:

- Define safety culture
- Discuss why safety culture is important
- Describe the safety culture cross-cutting aspects in the ROP
- Describe the treatment of safety culture in the ROP baseline and supplemental inspection programs

2. About Safety Culture

Safety Culture is the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.

The concept of safety culture is related to the concept of organizational culture. Organizational culture refers to the shared reactions and habits, values and norms, and beliefs and basic assumptions of the organization's members, and, in general, the way of action and way of solving problems in the organization (Schein, E.H., 2010).

Safety culture is:

- A subset of the organizational culture, including all factors which have or may have an impact on safety or security.
- The organizational culture seen from the safety point of view.

However, in relation to nuclear safety, the concept of safety culture has a specific meaning: it is used in the normative way as a requirement to maintain and support a strong and high-level safety culture in nuclear power plant sites, with the aim of ensuring safety in all operations.

2.1. Why Do We Care?

The accident at the Chernobyl nuclear power plant in 1986 brought attention to the importance of safety culture and the impact that weaknesses in safety culture can have on safety performance. Since then, the importance of a positive safety culture has been demonstrated by a number of significant, high-visibility events worldwide.

In the United States, assessments of various incidents involving civilian uses of radioactive materials have revealed that weaknesses in safety culture were either an underlying cause or increased the severity of the incidents. These events have not been confined to a particular type of licensee or certificate holder, as they have occurred at nuclear power plants and fuel cycle facilities and during medical and industrial activities involving radioactive materials. The causes of these incidents included, for example, inadequate management oversight of process changes, perceived production pressures, lack of a questioning attitude, and poor communications.

Below are three such incidents that demonstrate how safety culture weaknesses can contribute to poor performance and catastrophic events.

Chernobyl

On April 23, 1986, a catastrophic nuclear accident occurred at the Chernobyl Nuclear Power Plant in Ukraine. An explosion and fire released large quantities of radioactive particles into the atmosphere, which spread over much of the western Soviet Union and Europe.

The term "safety culture" was first used by the International Atomic Energy Agency (IAEA) to describe the human elements to the Chernobyl accident. Safety culture related causes included:

- Inadequate procedural adherence issues
- Non-conservative decisionmaking
- Lack of clear authority
- Poor training and understanding of the nuclear reactor technology
- Production (testing) over safety

Numerous IAEA follow-on activities were conducted and documents were developed to increase the understanding of safety culture.

Davis-Besse Reactor Vessel Head Degradation

On March 6, 2002, the Davis-Besse Nuclear Power Station found a football-sized cavity in the reactor pressure vessel head. The corrosion of the reactor vessel head was caused by boric acid leakage from the cracks in the control rod drive mechanism nozzle penetrations. Safety culture was a self-identified root cause, including:

- Less than adequate nuclear safety focus
- Less than adequate analysis of safety implications
- Inadequate corrective actions that addressed symptoms rather than root causes
- Failure to integrate and apply operating experience to plant conditions

The incident at Davis-Besse indicated the need for additional NRC efforts to evaluate whether the agency should increase its attention to reactor licensees' safety cultures. This resulted in important changes to the NRC's Reactor Oversight Process (ROP) to more fully address safety culture. Additional information on the incident can be found at http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html

Columbia Space Shuttle

On February 1, 2003, the Columbia space shuttle disintegrated as it reentered Earth's atmosphere, killing all seven crew members. The shuttle was damaged from a piece of foam insulation that broke off during launch and struck the left wing. NASA was aware of the foam shedding issue, and previous shuttle launches had experienced minor damage, but some engineers suspected that the damage to Columbia was more serious. NASA managers limited the investigation, believing that the foam shedding was a known issue and not a significant safety risk. The accident investigation identified multiple safety culture related causes, including:

- Reliance on past success as a substitute for sound engineering practices
- Lack of effective communication of safety information and stifled questioning attitude
- Lack of integrated management across program elements
- An informal chain of command that operated outside of the organization's formal rules

The Columbia Accident Investigation Board report highlights the importance of maintaining a questioning attitude toward safety, and the possible negative consequences that can occur when such a questioning attitude is lost or compromised. The report can be found at: http://www.nasa.gov/columbia/home/CAIB_Vol1.html

2.2. History

The history of safety culture is directly related to the changes and growth of both the nuclear industry and the NRC as an agency. Although it can be complicated, the basics of how safety culture evolved at the NRC correlates with the actions of the Commission to help make the industry and the use of nuclear energy in our country safer and more reliable.

1986: Safety Culture Becomes an International Topic

Safety culture became a major topic of discussion in the international community following the Chernobyl accident in 1986.

Safety culture related causes of the Chernobyl event were: Inadequate procedure adherence issues; non-conservative decisionmaking; lack of clear authority; poor training/understanding; and production (testing) over safety.

1989: NRC Policy Statement on Conduct of Operations

The concept of safety culture was first introduced in NRC policy documents in 1989, following reports to the NRC of instances of operator inattentiveness and unprofessional behavior in the control room of a nuclear power plant. This prompted the NRC to issue a policy statement on expectations for maintaining a professional work environment in the control room.

In the policy statement, the NRC described safety culture as "the necessary full attention to safety matters," and "the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants...Management has the duty and obligation to foster the development of a 'safety culture' at each facility and to provide a professional working environment, in the control room and throughout the facility, that assures safe operations."

1996: NRC Policy Statement on Freedom to Raise Safety Concerns Without Fear of Retaliation

In 1996 the NRC responded to an incident where workers were retaliated against for whistleblowing, resulting in guidance and a policy statement on expectations for maintaining a safety-conscious work environment, where workers can raise nuclear safety concerns without fear of retaliation. This policy statement applies to all NRC licensees and certificate holders.

2006: Davis Besse Event Prompts Changes in ROP to More Fully Address Safety Culture

The Davis Besse event in 2002 prompted a number of changes in NRC's oversight process. These changes, which were implemented in 2006, sought to:

- Provide opportunities to diagnose safety culture weaknesses and take action through enhancement of Inspection Procedure (IP) 71152 and the treatment of cross-cutting issues.
- Provide a process for determining the need to evaluate a licensee's safety culture through enhancement of IP 95002 and Inspection Manual Chapter (IMC) 0305.
- Provide structure guidance to evaluate a licensee's safety culture assessment and independently conduct an assessment through enhancement of IP 95003.

2011: Safety Culture Policy Statement

At the direction of the Commission, the NRC staff began an effort in 2008 to expand the Commission's safety culture policy to address the unique aspects of security and ensure applicability to all licensees and certificate holders. The NRC engaged in a collaborative effort with stakeholders, including Agreement States, to develop a definition of nuclear safety culture and a list of traits that describe a positive safety culture. The Final Safety Culture Policy Statement was approved by the Commission on March 7, 2011, and became effective upon publication in the Federal Register on June 14, 2011 (76 FR 34773).

2014: Safety Culture Common Language

Following publication of the NRC's Safety Culture Policy Statement, the NRC and the nuclear industry engaged in a joint effort to develop a common language around safety culture, using the traits in the policy statement as a starting framework. The result of the common language initiative were 10 traits of a healthy safety culture (the 9 traits from the policy statement, plus the addition of decisionmaking as a 10th trait), 40 aspects nested under those traits, and numerous examples for each aspect. In 2014, the common language traits and aspects were incorporated under the three cross-cutting areas of the Reactor Oversight Process.

2.3. Key Terms

It is important to learn the specific terms involving safety culture because some of them, like cross-cutting aspects, areas, and themes are obviously related but have very different meanings. For example, historically, there has been confusion between the terms "safety culture" and "safety conscious work environment (SCWE)." They are not the same thing and are not interchangeable terms. SCWE refers to an environment in which employees feel free to raise safety concerns whereas Safety Culture is a set of core values and behaviors that emphasize safety at the workplace. Thus a SCWE is actually an attribute of safety culture. More complete definitions of SCWE and safety culture are listed below.

Safety Culture

The core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.

Safety Conscious Work Environment

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.

Cross-Cutting Aspect (CCA)

The performance characteristic of an inspection finding that is either the primary cause of the performance deficiency or the most significant contributing cause.

Cross-Cutting Area

Fundamental performance attributes that extend across all of the Reactor Oversight Process (ROP) cornerstones of safety. The three cross cutting areas are:

- Human performance

- Problem identification and resolution
- SCWE

Cross-Cutting Theme
Multiple inspection findings with causes that share the same cross-cutting aspect during a single assessment period.

Substantive Cross-Cutting Issue (CCI)
A CCI is a cross-cutting theme which has been identified in at least three consecutive assessment letters.

3. Safety Culture Policy Statement
The NRC's Safety Culture Policy Statement emphasizes the importance the NRC places on the development and maintenance of a positive safety culture for all regulated activities. Policy statements help to guide the activities of the NRC staff and can express the Commission's expectations of others.

The Safety Culture Policy Statement sets forth the expectation that individuals and organizations performing NRC-regulated activities establish and maintain a positive safety culture commensurate with the safety and security significance of their actions. It applies to all licensees, certificate holders, permit holders, authorization holders, holders of quality assurance program approvals, vendors and suppliers of safety-related components, and applicants for a license, certificate, permit, authorization, or quality assurance program approval, subject to NRC authority.

3.1. Traits of a Positive Safety Culture
The Safety Culture Policy Statement includes a list of nine traits further defining a positive safety culture. These traits describe patterns of thinking, feeling, and behaving that emphasize safety, particularly in goal conflict situations, such as when safety goals conflict with production, schedule or cost goals.

Note that this list of traits is not all-inclusive and some organizations may find that one or more traits are particularly relevant to their activities. For example, as a result of the NRC's common language initiative with the nuclear power industry, decisionmaking was labeled an additional safety culture trait that is particularly important for nuclear power operations, and cross-cutting aspects specific to decisionmaking are included in the ROP.

Leadership Safety Values and Actions
Leaders demonstrate a commitment to safety in their decisions and behaviors.

Work Processes
The process of planning and controlling work activities is implemented so that safety is maintained.

Effective Safety Communications
Communications maintain a focus on safety.

Problem Identification and Resolution
Issues potentially impacting safety are promptly identified, fully evaluated, and promptly addressed and corrected commensurate with their significance.

Continuous Learning
Opportunities to learn about ways to ensure safety are sought out and implemented.

Respectful Work Environment
Trust and respect permeate the organization.

Decision Making
Decisions that support or affect nuclear safety are systematic, rigorous, and thorough.

Personal Accountability
All individuals take personal responsibility for safety.

Environment for Raising Concerns
A safety conscious work environment is maintained where personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment or discrimination.

Questioning Attitude
Individuals avoid complacency and continually challenge existing conditions and activities in order to identify discrepancies that might result in error or inappropriate action.

4. Cross-Cutting Areas
The NRC's ROP includes three cross-cutting areas, which are fundamental performance characteristics that extend across all of the ROP cornerstones of safety. These areas are human performance, problem identification and resolution (PI&R), and SCWE.

The safety culture cross-cutting aspects were developed and incorporated under the ROP cross-cutting areas in an effort to capture the important characteristics of safety culture which are observable to the NRC staff during inspections and assessments of licensee performance.

The NRC assigns cross-cutting aspects to inspection findings in accordance with IMC 0612, "Power Reactor Inspection Reports." The NRC reviews cross-cutting aspects for cross-cutting themes and potential CCIs in accordance with IMC 0305, "Operating Reactor Assessment Program," to provide licensees the opportunity to address performance issues before they result in more significant safety concerns. Although the presence of CCAs or the assignment of a CCI may be indicative of a potentially degraded safety culture, the NRC only draws conclusions about safety culture based on the results of licensee and NRC safety culture assessments conducted by qualified staff, not based on the presence of CCAs or CCIs.

The cross-cutting aspects and their definitions are currently provided in IMC 0310, "Aspects within the Cross-Cutting Areas," The definitions are listed in the next section of this module.

4.1. Cross-Cutting Areas and Aspects
Included in routine baseline inspections are twenty-three aspects that are divided into three cross-cutting areas. These areas are **Human Performance**, **Problem Identification and Resolution**, and **Safety Conscious Work Environment (SCWE)**. With the twenty-three aspects covered by the three main cross-cutting areas, there are also twelve additional aspects to be considered during supplemental and

reactive inspections which are intended to be reviewed more directly. Each of the aspects within each area is listed in the table below.

Also, the cross-cutting areas should be applied in the context of evaluating potential cross-cutting aspects of findings when determining the causes of performance deficiencies. The aspects should not be used as an inspection checklist. The aspects should not be used as an inspection checklist, but should only be applied when indicative of present performance.

Here are a couple of links to some helpful resources: NUREG-2165 and IMC-0310

Human Performance (H)

Resources [H.1]

Personnel, equipment, procedures, and other resources are available and adequate.

Field Presence [H.2]

Leaders are commonly seen in the work areas of the Deviations from standards and expectations are corrected promptly. Senior Managers ensure supervisory and management oversight of work activities.

Change Management [H.3]

Leaders use a systematic process for evaluating and implementing change.

Teamwork [H.4]

Individuals and work groups communicate and coordinate their activities.

Work Management [H.5]

The organization implements a process of planning, controlling, and executing work activities.

Design Margins [H.6]

The organization operates and maintains equipment within design margins. Special attention is placed on maintaining fission product barriers, defense-in-depth, and safety related equipment.

Documentation [H.7]

The organization creates and maintains complete, accurate, and up-to-date documentation. Procedure Adherence: Individuals follow processes, procedures and work instructions.

Procedure Adherence [H.8]

Individuals follow processes, procedures and work instructions.

Training [H.9]

The organization provides training and ensures knowledge transfer.

Bases for Decision [H.10]

Leaders ensure that the bases for operational and organizational decisions are communicated in a timely manner.

Challenge the Unknown [H.11]

Individuals stop when faced with uncertain conditions.

Avoid Complacency [H.12]

Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk.

Consistent Process [H.13]
Individuals use a consistent, systematic approach to make decisions.

Conservative Bias [H.14]
Individuals use decision making practices that emphasize prudent choices over those that are simply allowable.

Problem Identification and Resolution (P)

Identification [P.1]
The organization implements a corrective action program with a low threshold for identifying issues.

Evaluation [P.2]
The organization thoroughly evaluates issues to ensure that resolutions address the cause and extent of conditions.

Resolution [P.3]
The organization takes effective corrective actions to address issues.

Self-Assessment [P.4]
The organization routinely conducts self-critical and objective assessments.

Operating Experience [P.5]
The organization systematically and effectively collects, evaluates and implements relevant internal and external operating experience.

Trending [P.6]
The organization periodically analyzes information from the corrective action program and other assessments in the aggregate.

Safety Conscious Work Environment - SCWE (S)

SCWE Policy [S.1]
The organization effectively implements a policy that supports individuals' rights and responsibilities to raise safety concerns.

Alternate Process for Raising Concerns [S.2]
The organization effectively implements a process for raising and resolving concerns that is independent of line management influence.

Free Flow of Information [S.3]
Individuals communicate openly and candidly.

Supplemental Cross-Cutting Aspects (X)

Incentives, Sanctions and Rewards [X.1]
Leaders ensure incentives, sanctions and rewards are aligned with nuclear safety policies.

Strategic Commitment to Safety [X.2]
Leaders ensure plant priorities are aligned to reflect nuclear safety as the overriding priority.

Roles, Responsibilities, and Authorities [X.3]
Leaders clearly define roles, responsibilities, and authorities.

Constant Examination [X.4]
Leaders ensure that nuclear safety is constantly scrutinized.

Leader Behaviors [X.5]
Leaders exhibit behaviors that set the standards for safety.

Standards [X.6]
Individuals understand the importance of adherence to nuclear standards.

Job Ownership [X.7]
Individuals understand and demonstrate personal responsibility.

Benchmarking [X.8]
The organization learns from other organizations to continuously improve.

Work Process Communication [X.9]
Individuals incorporate safety communications in work activities.

Expectations [X.10]
Leaders frequently communicate and reinforce the expectation that nuclear safety is the organization's overriding priority.

Challenge Assumptions [X.11]
Individuals challenge assumptions and offer opposing views when they think something is not correct.

Accountability for Decisions [X.12]
Single-point accountability is maintained for nuclear safety decisions.

4.2. Cross-Cutting Themes and Issues

The NRC identifies cross-cutting issues (CCIs) to inform the licensee that the NRC has a concern with the licensee's performance in a cross-cutting area and to encourage the licensee to take appropriate actions before more significant performance issues emerge. A CCI is a cross-cutting theme which has been identified in at least three consecutive assessment letters. CCIs are identified on a "per site" basis; not on a "per unit" basis.

Cross-Cutting Themes

For the cross-cutting areas of problem identification and resolution (PI&R) and human performance (HU), a cross-cutting theme exists when at least six inspection findings are assigned the same cross-cutting aspect (CCA) during a mid-cycle or end-of-cycle assessment period. Note that the findings should be representative of more than one cornerstone; however, given the significant inspection effort applied to the Mitigating Systems Cornerstone, a cross-cutting theme can exist consisting of inspection findings associated with only this one cornerstone.

A cross-cutting theme exists in the area of safety conscious work environment (SCWE) if at least one of the following three conditions exists in an 18-month period (i.e., the current mid- or end-of-cycle assessment period and the two quarters preceding that period):

1. A finding with a documented CCA in SCWE and the impact on SCWE was not isolated, or
2. The licensee has received a chilling effect letter, or
3. The licensee has received correspondence from the NRC that transmitted an enforcement action with a Severity Level (SL) I, II, or III, and that involved discrimination, or a confirmatory order that involved discrimination.

A cross-cutting theme also exists if during the previous 12-month assessment period, a licensee has at least 20 findings with cross-cutting aspects in the Human Performance cross-cutting area, or 12 findings with cross-cutting aspects in the Problem Identification and Resolution cross-cutting area.

Documenting Cross-Cutting Themes and Opening Cross-Cutting Issues

The first time that a licensee meets the criteria for a cross-cutting theme, the region will document the theme in the assessment letter. The region should review licensee actions with regards to a causal analysis and/or corrective actions for that theme.

For the second consecutive assessment meeting with the same cross-cutting theme, the region will document the theme in the assessment letter again. If not already done, the region should consider the effectiveness of licensee actions (e.g., additional findings with the same aspect during the last six months of the assessment cycle) in determining whether or not to perform additional follow-up of licensee corrective actions. Regional follow-up of licensee corrective actions could be accomplished through a PI&R inspection sample, a semi-annual trend review focused on the theme, or including it within the scope of a biennial PI&R inspection, if one is scheduled during the period.

For the third consecutive assessment meeting with the same cross-cutting theme, the region will open and document a CCI in the assessment letter.

Additional information regarding opening and closing CCIs can be found in IMC 0305.

5. Safety Culture Assessment

The NRC's approach to safety culture assessment is a graded process.

The extent and complexity of a safety culture assessment is generally based on a licensee's placement in the ROP Action Matrix, and assessments may also be performed to follow-up on CCIs.

The scope and complexity increases with increased oversight and the focus of the assessment may be tailored based on the original performance deficiency—an assessment may focus more heavily on one part of the plant, or on one area of safety culture, like safety-conscious work environment.

The NRC can request a licensee to have an assessment of their safety culture performed if the licensee meets any of the following conditions:

Recurring Cross-Cutting Issue

The NRC may request that a licensee perform an assessment of their safety culture when the same cross-cutting issue has been identified in two or more consecutive assessment letters.

Degraded Cornerstone Column of the Action Matrix

The NRC may request that a licensee perform an independent assessment of their safety culture if the NRC identified through the conduct of Supplemental Inspection Procedure 95002, "Inspection for One

Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," and the licensee did not recognize, that one or more aspects of safety culture caused or significantly contributed to the risk significant performance issues. This will be discussed further later in the module.

Multiple/Repetitive Degraded Cornerstone Column of the Action Matrix

The NRC expects that a licensee in the Multiple/Repetitive Degraded Cornerstone Column of the Action Matrix will have a third party assessment of their safety culture performed.

5.1. 95002/95003 Inspection

The visible aspects of an organization's safety culture can be assessed by evaluating the extent to which its policies, programs, and processes ensure that nuclear safety issues receive the attention warranted by their significance. For example, the effectiveness of the licensee's corrective action program at identifying, prioritizing, and resolving issues with nuclear safety impacts provides important insights into the licensee's safety culture. An organization's members' shared attitudes and behaviors with respect to nuclear safety also provide important insights into a licensee's safety culture and can be assessed through behavioral observations, interviews, and focus groups.

IP 95002, "Inspection for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area"

As part of IP 95002, NRC inspectors perform a focused inspection to independently determine that the licensee's root cause evaluation appropriately considered whether any aspect of the licensee's safety culture caused or significantly contributed to any risk-significant performance issue. Activities associated with this inspection typically include:

- Reviewing the licensee's third party safety culture assessment.
- Reviewing the licensee's root cause evaluations to determine whether or not they identified which safety culture aspects led to the performance deficiencies.
- Conducting of a limited number of focus groups and interviews with site personnel to understand safety culture issues.

IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input"

The purpose of the safety culture portions of an IP 95003 inspection is for the NRC to evaluate the licensee's safety culture assessment and perform an independent assessment of the safety culture. Activities associated with this inspection typically include:

- Reviewing the licensee's third party safety culture assessment to determine the scope of the inspection.
- Performing a graded safety culture assessment by conducting focus groups and interviews with personnel from across the organization.
- Conducting behavioral observations of relevant plant activities and meetings to identify patterns of behavior related to safety culture.
- Conducting a document review of relevant procedures, training materials, files, etc. pertaining to safety culture and safety conscious work environment.

6. Inspection Manual Chapters and Procedures

Below are the descriptions of the different treatments of Safety Culture in the Baseline and Supplemental Inspection Programs across the agency. Each chapter or procedure overview has a link to the online document located on the Inspection Manual Public Web site to ensure currency of inspection guidance.

6.1. IMC 0612, "Issue Screening"

Purpose

The purpose of this procedure is to provide requirements for inspection issue screening and ensure that all violations of NRC requirements by power reactor licensees are appropriately dispositioned in accordance with the NRC Enforcement Policy.

Objectives

- Screen inspection results
- Ensure violations are dispositioned in accordance with NRC Enforcement Policy
- Ensure all deficiencies are screened, assessed and documented.

6.2. IMC 0305, "Operating Reactor Assessment Program"

Purpose

The purpose of this procedure is to integrate the NRC's inspection, assessment, and enforcement programs and to overall safety performance.

Objectives

- Collect information from inspection findings and performance indicators (PIs).
- Arrive at an objective assessment of licensee safety performance using PIs and inspection findings.
- Assist NRC management in making timely and predictable decisions regarding appropriate agency actions used to oversee, inspect, and assess licensee performance.
- Provide a method for informing the public and soliciting stakeholder feedback on the NRC's assessment of licensee performance.
- Provide a process to follow up on areas of concern.

Treatment of Safety Culture

- Provides guidance on the cross-cutting areas, which are comprised of safety culture aspects.
- IMC 0310 contains definitions of safety culture and the safety culture aspects.
- Offers options, with specific criteria, to allow the NRC to request a licensee to have an assessment of their safety culture performed.

6.3. IP 71152, "Identification and Resolution of Problems"

Purpose

The purpose of this procedure is to assess a licensee's problem identification and resolution, accomplished through:

- Daily, routine review
- Quarterly samples
- Semiannual trend reviews
- Biennial team inspection

Treatment of Safety Culture

In Inspection requirements, directs inspectors to:

- Be aware of safety culture aspects.
- Inspect and assess corrective actions program, licensee use of operating experience, and licensee self-assessments and audits.
- Review a licensee's self-assessment of safety culture.

In Inspection Guidance, directs inspectors to:

- Include inspection of samples of operating experience and self-assessments and audits.
- Conduct samples of self-assessments and audits and alternate processes for raising concerns.
- Conduct review of a self-assessment of safety culture.

In Inspection Guidance, provides for the inspectors:

- Performance attributes for treatment of operating experience and effective self-assessments.
- Documentation instructions to address all of problem identification and resolution.
- Description of problems that may impact a SCWE.
- Sample questions for assessing SCWE.

6.4. IP 93100, "Safety Conscious Work Environment Concern Follow-up"

Purpose

- To inspect the safety-conscious work environment (SCWE) attribute of a licensee's safety culture. Insights gathered during this inspection would be considered during the mid- or end-of-cycle assessment meetings.
- The SCWE cross-cutting area is sampled during the biennial problem identification and resolution team inspection.
- When directed by management, SCWE-related issues of concern identified during IP 71152 can be examined in more depth.

Objectives

- Determine if indications of a chilled work environment exist.
- Determine if employees are reluctant to raise safety or regulatory issues
- Determine if employees are being discouraged from raising safety or regulatory issues

6.5. IP 40100, "Independent Safety Culture Assessment Follow-up"

Purpose

Purpose of the inspection procedure:

- Provides guidance for following up on a U.S. Nuclear Regulatory Commission (NRC) request for a licensee to perform an independent safety culture assessment.
- The NRC can ask a licensee to perform an independent safety culture assessment for the following situations:
    - a conclusion is reached during an inspection that the licensee did not adequately evaluate the contribution of a safety culture trait to the performance issue, or
    - a licensee has not adequately addressed a repetitive substantive cross-cutting issue (SCCI), which may be indicative of underlying organizational issues with safety culture implications.

6.6. IP 71153, "Follow-up of Events and Notices of Enforcement Discretion"

Purpose

The purpose of this procedure is to provide inspector response to site and Licensee Event Report (LER) reviews.

Treatment of Safety Culture

In Inspection Requirements, directs inspectors to:

- Retain observations related to apparent performance issues and contributing factors

In Inspection Guidance, directs inspectors to:

- Provide any information on potential contributing factors that may assist the follow-up assessment
- Include any issues noted with aspects of safety culture
- Information is provided for follow-up by IIT, AIT, SI, or ROP inspection(s).

6.7. IP 93800, "Augmented Inspection Team"

Purpose

The purpose of this procedure is to review an event with a larger, more experienced team based on the significance of the event.

Treatment of Safety Culture

In Inspection Guidance, directs inspectors to emphasize fact finding including:

- Conditions preceding the event
- Chronology
- Systems response
- Equipment performance
- Precursors
- Human factors considerations
- Quality assurance considerations
- Radiological considerations
- Safeguards considerations
- Safety culture aspect considerations

In Inspection Documentation, directs inspectors to:

- Document probable contributing causes of the event or degraded condition related to the safety culture aspects.
- Due to the sensitive nature of AITs, areas where no findings are identified should be documented in greater detail than required
- The results of this inspection may be used to inform a subsequent supplemental inspection (IP 95001, IP 95002, or IP 95003) based on the final significance determination of any findings associated with the event.
- The AIT leader should provide any information on potential causes or contributing factors, including safety culture issues, to the team leader of any related supplemental inspection.

6.8. IP 93812, "Special Inspection"

Purpose

The purpose of this procedure is to provide guidance to assess an event and its causes with a special inspection team.

Treatment of Safety Culture

In Inspection Guidance, directs inspectors to emphasize fact finding including:

- Conditions preceding the event
- Chronology
- Systems response
- Equipment performance
- Precursors
- Human factors considerations
- Quality assurance considerations
- Radiological considerations
- Safeguards considerations
- Safety culture aspect considerations

In Inspection Documentation, directs inspectors to:

- Document probable contributing causes of the event or degraded condition related to the safety culture aspects.
- Due to the sensitive nature of SIs, areas where no findings are identified should be documented in greater detail than required
- The results of this inspection may be used to inform a subsequent supplemental inspection (IP 95001, IP 95002, or IP 95003) based on the final significance determination of any findings associated with the event.
- The SI leader should provide any information on potential causes or contributing factors, including safety culture issues, to the team leader of any related supplemental inspection.

6.9. IP 95001, “Inspection for One or Two White Inputs in a Strategic Performance Area”

Purpose

Purpose is to provide assurance that:

- The root causes and contributing causes of risk significant performance issues are understood.
- The extent-of-condition and extent-of-cause of risk significant performance issues are identified.
- Licensee corrective actions to risk significant performance issues are sufficient to address the root and contributing causes, and to prevent recurrence.

Treatment of Safety Culture

In Inspection Requirement, directs inspectors to:

- Determine that the root cause evaluation, extent-of-condition, and extent-of-cause appropriately considered the safety culture aspects described in IMC 0310.

In Inspection Guidance, directs inspectors to:

- Determine whether a weakness in a safety culture aspect was a root cause or contributing cause.
- If so, verify the licensee addressed that weakness through appropriate corrective actions.
- If a weakness in a safety culture aspect was a root cause or contributing cause AND the licensee did not recognize and address that cause, this is a weakness in their evaluation, and the licensee may be subject to additional agency actions.

6.10. IP 95002, “Inspection for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area”

Purpose

- Provide assurance that the root causes and contributing causes are understood
- Assess the extent-of-condition for individual and collective risk significant performance issues
- Provide assurance that licensee corrective actions to risk significant performance issues are sufficient

Treatment of Safety Culture

In inspection Objectives, directs inspectors to:

- Independently determine whether any safety culture aspect caused or contributed significantly to risk significant performance issues.

In Inspection Requirements, directs inspectors to:

- Determine that the root cause evaluation appropriately considered safety culture aspects.
- Independently perform an evaluation.
- NRC may request that the licensee complete an independent assessment of safety culture.

In Inspection Guidance, provides:

- Guidance for making the determination required above.
- A note that failure to consider a safety culture aspect is not necessarily a violation.

6.11. IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input"

Purpose

- Provide the NRC additional information
- Provide an independent assessment
- Independently assess the adequacy of the programs and processes
- Independently evaluate the adequacy of programs and processes
- Provide insight into the overall root and contributing causes of deficiencies
- Determine if the NRC oversight process provided sufficient warning
- Independently assess the licensee's safety culture and evaluate the licensee's assessment of its safety culture.

Treatment of Safety Culture

Contains a boundary condition to be met before inspection begins:

- The licensee has completed a third party assessment of their safety culture.

In Inspection Requirements, directs inspectors to:

- Perform a review of the licensee's third party safety culture assessment
- Conduct the NRC's independent assessment of safety culture

IP 95003 inspection teams will receive "just-in-time" training before performing the inspection.

6.12. IP 95003.02, "Guidance for Conducting an Independent NRC Safety Culture Assessment"

Purpose

The purpose of this attachment is to provide guidance for the NRC to assess a licensee's safety culture. The guidance includes:

- Preparation for an independent safety culture assessment
- Conducting an independent safety culture assessment
- Sample inspection requirements for safety culture traits
- Sample questions for safety culture traits
- Guidance for focus groups and individual interviews
- Guidance for structured behavioral observations
- Guidance for safety culture event follow-up studies
- Guidance for evaluating safety culture surveys

7. Case Studies

This section of the training module includes 4 case studies to help you understand how cross-cutting aspects are assigned to inspection findings. Each case includes a narrative, description of the associated inspection finding, followed by causes and significant contributors to the incident, and finally the cross-cutting aspect assigned to the inspection finding. Pay close attention to each case narrative and try to independently determine which cross-cutting aspect best fits the situation.

7.1. Case Study 1

On May 10, 2011, following a refueling outage, the reactor mode switch was taken to startup at 06:26, and control rod withdrawal commenced at 06:41. The control room crew consisted of the following personnel (additional licensed operators were present in the control room conducting various startup related activities):

- Assistant Operations Manager (AOM-Shift) - Senior Line Management oversight
- Shift Manager (SM) - management oversight
- Reactivity Senior Reactor Operator/Control Room Supervisor (SRO/CRS) - command and control
- Assistant Control Room Supervisor (ACRS)
- Reactor Operator At-The-Controls (RO-ATC)
- Reactor Operator (Verifier) - ATC verifier
- Reactor Engineer (RE)
- Reactor Engineer in Training (RE in Training)

At 12:12, the reactor was made critical. Power continued to rise to the point of adding heat (POAH), and the POAH was achieved at 12:27. At approximately 12:31, the Reactivity SRO/CRS and the RO-ATC were relieved by other licensed operators who continued with plant startup. The crew withdrew control rods to establish a moderator heat-up rate. The RO-ATC withdrew four control rods without incident, then tried to withdraw control rod 30-11 from position 8 to 12, but the control rod would not move using normal notch withdraw commands. The RO-ATC then attempted to use a "double-clutch" maneuver in accordance with procedures; however, the control rod inadvertently inserted and settled at position 6.

The RO-ATC, the ATC verifier, and the Reactivity SRO/CRS all saw the control rod in the incorrect position. However, the operators did not enter and follow Procedure 2.4.11, "Control Rod Positioning Malfunctions" as required. This procedure required the operators to assess the mispositioning to determine the appropriate course of action before proceeding, and required the issue be documented in a condition report. During interviews, the three operators each indicated that there was confusion in their mind regarding whether or not the control rod met the definition of a mispositioned control rod because the control rod was only out of position by one notch from the initial position, but none of the operators referred to the procedure, and there was no discussion or challenge regarding the proper course of action among the operators. The condition was not logged, and a condition report was not generated until the issue was identified by NRC inspectors.

Following withdrawal of the five control rods (ten control rod notches), the RO-ATC observed the process computer displaying a high short-term (five minute average) moderator heat-up rate reading of 18°F per 5 minutes that he mistakenly believed corresponded to an hourly heat-up rate of 216°F/hr (the actual hourly heatup rate was 50°F/hr). The heat-up rate concern was discussed among the SM, Reactivity SRO/CRS, RO-ATC, Verifier and AOM-Shift. After the discussion, the SM directed the crew at the controls to insert control rods to reduce the heat-up rate. This direction did not include specific guidance or limitations regarding the number of control rod notches to insert. At this point, the AOM-Shift and SM left the front panels area of the control room.

The RE and RE-in-training were at computer terminals in the control room performing required calculations related to startup. The RE in training overheard the operator conversation about inserting control rods. He informed the RE, who in turn, questioned the SM about the decision to insert rods. The SM responded that the actions were necessary to control heat-up rate, and no further discussion occurred. During interviews with the NRC inspectors, the SM and the AOM-Shift stated that they both discussed that there was a need to be careful to avoid taking the reactor subcritical and that the action of inserting control rods had the potential to cause the reactor to become subcritical. However, this important information was never communicated to any of the operators at the controls.

As a result of the previous control rod withdrawal, moderator temperature was 40°F higher than it was at initial criticality, resulting in slightly increased control rod worth. The crew did not factor this increased control rod worth into their decision regarding the number of control rod notches to insert. Over the next three minutes, the RO-ATC proceeded to re-insert control rods that had been previously withdrawn to establish the heat-up. At the end of the rod insertion evolution, the SM directed the Reactivity SRO/CRS and the RO-ATC operator to keep reactor power on IRM range 7. This communication was not acknowledged by the RO-ATC. During interviews with the NRC inspectors, none of the operators recalled receiving such instructions. The SM then left the control room to take a break. The AOM-Shift left the controls area to get lunch in the control room kitchen.

As a result of the control rod insertions, reactor power lowered, thus requiring the RO-ATC to range the IRMs down to range 7 and then to range 6. The reactor had become subcritical, but the crew did not recognize the change in reactor status.

Approximately four minutes after the control rods were inserted to reduce the heat-up rate, the RO-ATC observed the process computer displaying a 0°F/hr heat-up rate. At this time, the SRO who had previously been relieved, returned and re-assumed his role as Reactivity SRO/CRS. The Reactivity SRO/CRS and the RO-ATC decided to once again withdraw control rods to re-establish the desired heat-up rate. Three of the same control rods were withdrawn, resulting in a rising IRM count rate that was observed by the operators. However, the crew did not recognize that the reactor status had changed from subcritical to critical.

At this point, the AOM-Shift returned to the reactor panel area. The RO-ATC continued rod withdrawal. The RO-ATC and the Verifier ranged the IRMs up as reactor power increased. The operators did not recognize the increasing rate of change in IRM power. Finally, the RO-ATC selected and withdrew control rod 30-11 from position 8 to 10. At 13:18, IRM readings rose sharply and an IRM Hi-Hi flux condition was experienced on both Reactor Protection System (RPS) channels, resulting in an automatic reactor scram.

Inspection Finding

A self-revealing finding was identified involving the failure of licensee personnel to implement conduct of operations and reactivity control standards and procedures during a reactor startup, which contributed to an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram. The significance of the finding was determined to be White, or of low to moderate safety significance. The finding was also associated with an apparent violation of NRC requirements specified by Technical Specification 5.4, "Procedures."

Causes and Significant Contributors

The inspection team determined that multiple factors contributed to the performance deficiency, including: inadequate enforcement of operating standards, failure to follow procedures, and ineffective operator training. The licensee's root cause evaluation determined that the primary cause was a failure to adhere to established standards and expectations due to a lack of consistent supervisory and management enforcement.

In addition, the licensee's root cause evaluation specified a number of condition reports and self-assessment reports written in the months preceding this event that demonstrated that the performance deficiency existed over an extended period of time and affected all operating crews. While the performance deficiency manifested itself during this particular low power event, there was the potential for the performance deficiency to result in a more consequential event under different circumstances.

Cross-Cutting Aspect

The inspection team concluded that the finding had a cross-cutting aspect of Avoid Complacency in the Human Performance cross-cutting area, because the licensee did not adequately enforce human error prevention techniques, such as procedural adherence, holding pre-job briefs, self and peer checking, and proper documentation of activities during a reactor startup, which is a risk significant evolution. Additionally, licensed personnel did not effectively implement the human performance prevention techniques mentioned above, and they did not recognize or plan for the possibility of mistakes, latent issues, or inherent risk during the reactor startup [H.12].

Avoid Complacency [H.12]: ***Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reduction tools.***

7.2. Case Study 2

During a refueling outage on March 31, 2013, a temporary overhead crane being used to move the generator stator from Unit 1 collapsed, causing the 525-ton stator to fall on and extensively damage portions of the turbine deck and then fall over 30 feet into the train bay. One worker was killed and eight others were injured.

The stator drop resulted in a Unit 1 loss of offsite power for 6 days and a Unit 2 reactor trip and loss of offsite power to one vital bus. The dropped stator ruptured a common fire main header in the train bay, which caused flooding in Unit 1 and water damage to the electrical switchgear for Unit 2. The alternate alternating current diesel generator (station blackout) electrical supply cables to both units were pulled out of the electrical switchgear and the diesel was therefore not available to either unit.

Inspection Finding

A self-revealing apparent violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified. It states, in part, that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings." The licensee did not follow the requirements specified in Procedure EN-MA-119, "Material Handling Program," in that, the licensee did not perform an adequate review of the subcontractor's lifting rig design calculation and the licensee failed to conduct a load test of the lifting rig prior to use.

The inspectors determined that the finding was more than minor because it was associated with the procedural control attribute of the initiating event cornerstone, and adversely affected the cornerstone's objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations.

Causes & Significant Contributors

The licensee concluded that one of the root causes for the temporary lift assembly collapse was that the sub-contractor's design did not ensure that the lift assembly north tower could support the loads anticipated for the lift. In addition, the licensee, based on its root cause evaluation, concluded that the subcontractor failed to conduct the required load testing of their modified temporary lift assembly before its use.

The inspectors determined that the cause for the stator drop was not following a quality-related procedure, in that, the overhead temporary hosting assembly was not properly designed; the associated calculation was not reviewed; and the assembly was not load tested as required. Further, the licensee did not provide a sufficient level of oversight in that, the requirements in Procedure EN-MA-119, for design approval and load testing of the temporary hoisting assembly, were not followed.

Cross Cutting Aspect(s)

This finding had a cross-cutting aspect in the cross-cutting area of human performance associated with Field Presence, because the licensee did not ensure adequate supervisory and management oversight of work activities, including contractors and supplemental personnel [H.2].

Field Presence [H.2]: Leaders are commonly seen in the work areas of the plant observing, coaching, and reinforcing standards and expectations. Deviations from standards and expectations are corrected promptly. Senior managers ensure supervisory and management oversight of work activities, including contractors and supplemental personnel.

7.3. Case Study 3

In 2014, during a maintenance rule sample and historical review of the 120 VAC vital instrument power (Y) system, inspectors identified a trend going back to 2008 of time delay relays failing in the red and blue channel instrument bus inverters.

The licensee has three red and three blue channel instrument bus inverters, which utilize identical relays for two separate functions; one relay location within each the inverter is designated as K1 and functions to initiate an alarm in the control room, while the other relay location is designated as K2 and functions to transfer the inverter to its non-safety related power supply after a loss of its safety-related supply. The K2 relay is continuously energized and its failure causes the associated inverter to unnecessarily transfer to its non-safety related backup power source and results in the inoperability of the inverter. The licensee's analysis of the failed K2 relays sampled three of the four failed relays and found that the same internal components of the relays failed. The licensee's analysis stated that the relays are continually energized and failure occurs when the board components burn out.

The inspectors found that the licensee performed a condition evaluation after the 2009 K2 relay failure and the evaluation concluded that the K2 relays should have been replaced at a 10-year frequency. Further review by the inspectors, found that two failures had occurred prior to the 2009 failure. Specifically, one K2 relay had failed in 2002 and one K1 relay had failed in 2008. Since the 2009 relay failure, the inspectors found that the inverters have had six K1 or K2 relay failures, all of which were the original relays from the 1991 inverter installations. The inspectors found that the condition evaluation corrective actions from the 2009 relay failure added the relay replacements to the scheduled 10-year inverter overhauls, but failed to evaluate if the remaining K2 relays, which were already 8 years past their prescribed replacement date, could remain in-service until the scheduled overhauls, which in some cases took the relays out to 23 years of in-service time.

Inspection Finding

A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was self-revealed for the failure to replace safety-related inverter components at the vendor prescribed 10-year frequency. Specifically, after concluding that safety-related inverter relays were required to be replaced at a 10-year frequency, per vendor direction, the licensee failed to promptly replace the remaining relays that were eighteen years old or evaluate if the relays could remain in-service until the next scheduled 10-year inverter overhaul.

Causes & Significant Contributions

After identifying that installed inverter relays were past their vendor prescribed 10-year replacement frequency, the licensee failed to promptly replace the remaining relays or evaluate if the relays could remain in-service until the next scheduled 10-year inverter overhaul. The inspectors determined that the finding was reflective of current performance because the licensee had multiple opportunities to implement effective corrective actions with each subsequent relay failure in 2010, 2012, 2013, and 2014.

Cross-Cutting Aspect(s)

This finding has a cross-cutting aspect of Resolution [P.3], in the cross-cutting area of Problem Identification and Resolution because the licensee failed to take effective corrective actions to address issues in a timely manner commensurate with their safety significance.

Resolution [P.3]: The organization takes effective corrective actions to address issues in a timely manner commensurate with their safety significance.

7.4. Case Study 4

At approximately 12:30 p.m. on May 24, 2014, during power ascension to 100 percent following a recent refueling outage, the plant stopped power ascension at approximately 91 percent after a SW leak on the main generator exciter air cooler occurred. At approximately 1:47 p.m., operators began a load reduction of approximately 10 percent per hour, and at approximately 2:00 p.m., station personnel determined that the SW leak was coming from an extruded gasket on the generator exciter air cooler reversing head and attempted to re-torque the head with no effect. At approximately 2:16 p.m., operators began a rapid load reduction of approximately 1 percent per minute, and the reactor was shut down at 3:56 p.m.

The licensee completed a root cause evaluation in order to determine the root and contributing causes and to propose corrective actions. The licensee identified two root causes—the cooler reversing head flatness was outside of specification, which caused excessive stresses on the gasket and resulted in localized compression failures; and the lack of clear guidance and acceptance criteria resulted in knowledge-based decision making that was predicated on successful historical performance.

During interviews and reviews of the root cause evaluation, the inspectors noted that the exciter air cooler work package (WO C92106668) required an inspection of the cooler head surfaces to determine the need to perform machining. Work planning and execution in the turbine services organization had been historically performed by the original equipment manufacturer (OEM) using the OEM's procedures and processes. The licensee's work package did not give any guidance on what parameters to check or any acceptance criteria to evaluate the as-found condition.

The vendor recommended performing machining on the cooler heads to repair the surface finish during the 2014 outage, but the reversing head was not sent for machining, the flatness was not checked on either head, and no CR was initiated by the licensee. Similar recommendations were made by the vendor during previous outages. Neither the 2009, 2012, nor the 2014 recommendations to machine the cooler heads were completed; and CRs were not initiated in 2009, 2012, or 2014 to document these recommendations and deficiencies.

Inspection Finding

A self-revealing Green finding was identified for inadequate development and maintenance of work packages as required by the licensee's procedure CNG-MN-4.01-1003, "Work Order Planning," Revision 00701. Specifically, the work packages associated with maintenance on the main generator exciter air cooler reversing head did not adequately incorporate and comply with vendor recommendations, which resulted in a service water (SW) leak on the reversing chamber of the generator exciter air cooler, a rapid downpower, and shutdown of the reactor.

Causes & Significant Contributions

The licensee did not initiate condition reports and document reversing head material deficiencies identified by the licensee's vendor and recommended for repair in 2009, 2012, and 2014.

Cross-Cutting Aspect(s)

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Identification, because the licensee did not implement a corrective action program with a low threshold for identifying issues, and individuals did not identify issues completely, accurately, and in a timely manner in accordance with the program [P.1].

Identification [P.1]: The organization implements a corrective action program with a low threshold for identifying issues. Individuals identify issues completely, accurately, and in a timely manner in accordance with the program.

8. Additional Resources

Here are a few resources that can assist you with more information about understanding Safety Culture within the NRC.

NRC's Safety Culture Webpage

This is the primary resource for safety culture information on the NRC's public website. It includes extensive background on the NRC's Safety Culture Policy Statement, and educational materials to help NRC staff, licensees, and other stakeholders learn about the role of safety culture in ensuring nuclear safety.

[NUREG-2165, “Safety Culture Common Language”]

This report presents a suggested common language, agreed upon by NRC staff and the nuclear industry, for classifying and grouping traits and attributes of a healthy nuclear safety culture. This report is the basis for the current cross-cutting aspects in the ROP and includes examples that can help inspectors determine which cross-cutting aspect best applies to an inspection finding.

[IMC-1245, Appendix C-12, “Safety Culture Assessor Training and Qualification Journal”]

This is the qualification standard for safety culture assessors developed by the Office of Nuclear Reactor Regulation (NRR). This standard outlines the training requirements and competencies needed to perform safety culture assessments at a licensee or vendor facility in accordance with inspection procedures (e.g., IP 95002 or IP 95003).

9. Contacts

For questions about this training, safety culture assessments, or the safety culture assessor qualification card, please contact:

Molly Keefe-Forsyth
Human Factors Specialist
ROP Assessment Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
Molly.Keefe@nrc.gov
301-415-5717

For questions about cross-cutting issues (CCIs), please contact:

Dan Merzke
Senior Reactor Operations Engineer
ROP Assessment Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
Dan.Merzke@nrc.gov
301-415-1457

For questions about the Safety Culture Policy Statement, please contact:

Diane Sieracki
Senior Safety Culture Program Manager
Concerns Resolution Branch
Office of Enforcement
Diane.Sieracki@nrc.gov
301-415-3297