Appendix C-2
Reactor Engineering Inspector
Technical Proficiency
Training and Qualification Journal

Table of Contents

Introduction

Do not begin the activities or complete the courses in this qualification journal until you have completed the Basic Inspector Certification Journal. You may complete the General Proficiency requirements contained in Appendix B together with the Technical Proficiency requirements outlined in this journal.

Before signing up for any course, be sure that you have checked and have met any prerequisites.

Required Reactor Engineering Inspector Training Courses

The following courses require completion of Appendix A as a prerequisite:

- Reactor Full Series (either boiling-water reactor (BWR) or pressurized-water reactor (PWR))
  - — BWR Series = R-304B, R-504B, and R-624B or
  - — PWR Series = R-304P, R-504P, and R-624P

- Basic Reactor Operations for alternate reactor type
  - — R-104P, if you completed the BWR series
  - — R-105B, if you completed the PWR series

The following course DOES NOT require the completion of Appendix A, but you must meet course prerequisites:

- Power Plant Engineering (self-study)

Post-Qualification Engineering Inspector Training Course

The following course IS NOT required for initial qualification; instead, it is a post-qualification requirement to be completed with 24 months of full qualification:

- Probabilistic Risk Assessment Technology and Regulatory Perspectives (P-111)

Required Refresher Training

One of the following is required once every 3 years:

- BWR or PWR systems refresher OR
- simulator refresher OR
- emergency operating procedure refresher

If you completed your qualification before the U.S. Nuclear Regulatory Commission (NRC) added the requirement for a full series, you should alternate between R-104B and R-104P. You should coordinate your selection with your supervisor.

Engineering Individual Study Guide

Engineering Individual Study Guide

TOPIC: (ISA-ENG-1) Capability of Safety Systems to Perform Intended Safety Functions

PURPOSE: The purpose of this guide is to acquaint you with the actions taken by the NRC in the review of safety systems to determine their capability to perform their intended safety function(s) and to discover any performance issues that hinder that capability. As a reactor engineering inspector, you will be required to understand how the inability of one or more systems to perform as intended causes increased risk for core damage and increased likelihood that the plant's inherent redundancy may not be able to mitigate the loss of the safety functions of those systems.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 32 hours

REFERENCES:

1. Inspection Procedure (IP) 71111.21, "Safety System Design and Performance Capability"
2. IP 93801, "Safety System Functional Inspection"
3. NUREG-1275, Volume 14, "Causes and Significance of Design-Based Issues at U.S. Nuclear Power Plants"
4. NUREG/CR-5640, "Overview and Comparison of U.S. Commercial Nuclear Power Plants"
5. Regulatory Guide (RG) 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Basis," December 2000 (ADAMS Accession No. ML003754825)
6. Nuclear Energy Institute (NEI) 97-04, "Design Basis Program Guidelines," Appendix B (ADAMS Accession No. ML003771698)
7. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants"
8. Inspection Manual Chapter (IMC) 2515
9. Part 9900 10 CFR Guidance, "10 CFR 50.59 Changes, Tests, and Experiments"

10. NRC staff safety evaluation report for a specific plant for the original operating license

11. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”

EVALUATION CRITERIA:

Upon completion of the tasks in this guide, you will be asked to demonstrate your understanding of the NRC’s periodic review of a sample of safety systems at a given nuclear power plant and the NRC’s continuing role in determining system performance deficiencies that impact a licensee’s quality assurance program by performing the following:

1. State the NRC’s inspection objectives for the reviews of samples of safety systems on a periodic basis and the reason for their importance.

2. Discuss the purpose of a safety system design inspection (SSDI) in terms of the following:
   a. overall objective for each inspection
   b. number of inspectors and their areas of expertise
   c. duration of each inspection and the allocated resources
   d. how the inspection is risk informed

3. State the basis for selecting systems for an SSDI and the various methods for that selection process. Emphasize some measure that can analytically determine or approximate the loss of a safety function.

4. Define the following terms and state how samples of each are developed and assembled/paired with others for review. State whether the reviews of these samples are considered a “vertical slice” inspection or if the inspection has now become a horizontal inspection (scope of inspection expanded):

   a. inspection attributes
   b. system needs
   c. system condition and capability
   d. inspection activity
   e. component inspection activity

5. State the factors that cause a “vertical slice” inspection to become a horizontal inspection and whether the original “vertical slice” inspection can be resumed.

6. Define the contribution of each of the following documents to an SSDI, and the benefit of each in determining the functional capability of one or more systems:

a. design-basis documents
b. licensing-basis documents
c. calculations and analyses
d. technical specifications
e. design changes and modifications
f. operator training manual
g. maintenance procedures
h. surveillance and inservice test procedure results
i. applicable vendor manuals
j. generic communications (information notices, bulletins, and generic letters)

7. For the listed documents in the previous question, state how each provides insights into the assessment of a licensee's quality assurance program, at a minimum with regard to the design and functional capability of safety systems.

8. Develop a set of criteria that establishes a reasonable likelihood about the functional and operational capability of safety systems. For that set of criteria, determine subsets of the minimum criteria that must be present to derive a similar reasonable affirmation.

9. Define the Reactor Oversight Program cornerstones that are verified by the reviews of safety systems via an SSDI.

TASKS:

1. Read the references in sufficient detail to perform adequately in accordance with the requirements of the evaluation criteria.

2. Meet with your supervisor, or the person designated to be your resource for this activity, and discuss the answers to the questions listed under the evaluation criteria.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item ISA-ENG-1

Engineering Individual Study Guide

TOPIC: (ISA-ENG-2) The NRC's Review of Temporary and Permanent Plant Modifications

PURPOSE: The purpose of this activity is to acquaint you with the actions taken by the NRC in the review of both temporary and permanent plant modifications of power reactor facilities. As a reactor engineering inspector, you will be required to understand how design changes resulting in hardware modifications or different operating requirements of a facility can potentially impact the plant's design and licensing basis, as well as the performance capability of safety systems and components.

COMPETENCY AREAS: INSPECTION

LEVEL OF EFFORT: 32 hours

REFERENCES:

1. NUREG-1397, "An Assessment of Design Control Practices and Design Reconstitution Programs in the Nuclear Power Industry"
2. IP 71111.17, "Permanent Plant Modifications"
3. IP 71111.23, "Temporary Plant Modifications"

EVALUATION CRITERIA: Upon completion of the tasks in this guide, you will be asked to demonstrate your understanding of permanent and temporary plant modifications and the NRC's continuing role in monitoring design changes to power reactor facilities through the review of these types of modifications by successfully performing the following:

1. State the NRC's inspection objectives for the reviews of both permanent and temporary plant modifications and indicate why they are important.
2. Discuss the typical format of both permanent and temporary modifications (including the major sections and the purpose of each).
3. Discuss how licensees control modifications both before and after implementation, including affected design documents and plant procedures.
4. Define the following terms:
   a. configuration management

b. current licensing basis
c. design
d. design basis
e. design-basis document
f. design change
g. design control
h. design margin
i. design output
j. engineering design bases
k. essential design documents
l. fully documented and auditable design

5. Justify why the NRC is concerned about agreement between the design change of a modification and the safety evaluation contained in the modification package. Be able to address the outside design basis and requirements for a license amendment for a design change.

6. State at least five of the types of changes that comprise the category "permanent plant modifications" and the reason for the inclusion of each one.

7. State at least five of the types of changes that comprise the category "temporary plant modifications" and the reason for the inclusion of each one.

8. State which Reactor Oversight Program cornerstones are verified by the independent reviews of permanent and temporary plant modifications.

9. List the following:

   a. types of design documents that may be affected by modifications
   b. types of plant procedures that could be affected by modifications

TASKS:

1. Read the references in sufficient detail to perform adequately in accordance with the requirements of the evaluation criteria.

2. Meet with your supervisor, or the person designated to be your resource for this activity, and discuss the answers to the questions listed under the evaluation criteria.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item ISA-ENG-2

Engineering Individual Study Activity

TOPIC: (ISA-ENG-3) Evaluations of Changes, Tests, and Experiments (10 CFR 50.59)

PURPOSE: The purpose of this activity is to acquaint you with how to review safety evaluations that are used to determine if the power reactor facility change, test, or experiment requires NRC approval before implementation. As a reactor engineering inspector, you will be required to understand how design changes resulting in hardware modifications or different operating requirements of a facility can potentially impact the plant's design and licensing basis, as well as the performance capability of safety systems and components.

The purpose of this activity is to do the following:

1. Familiarize you with the NRC regulations governing changes, tests, and experiments for commercial nuclear power facilities.

2. Enable you to demonstrate an ability to conduct an inspection under 10 CFR 50.59, "Changes, Tests, and Experiments," in accordance with IP 71111.02.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 24 hours

REFERENCES:

1. 10 CFR 50.59, "Changes, Tests, and Experiments"

2. RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," November 2000

3. NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation"

4. Inspection Manual Part 9900, 10 CFR Guidance 50.59.CFR

5. Regulatory Issue Summary (RIS) 2001-03, "Changes, Tests, and Experiments"

6. RIS 2001-09, "Control of Hazard Barriers" (guidance on the applicability of 10 CFR 50.59 to barriers)

7. Current regional or office guidance for processing potential violations of 10 CFR 50.59

8. IP 71111.02, “Evaluations of Changes, Tests, or Experiments”

9. IP 71152, “Identification and Resolution of Problems”

10. IP 71111.15, “Operability Evaluations”

EVALUATION CRITERIA: At the completion of this activity, you should be able to do the following:

1. State the criteria for when the licensee may make changes to the facility or procedures or perform tests or experiments without obtaining prior NRC approval.

2. State the meaning of key terms used in this regulation—updated final safety analysis report (UFSAR), changes, facility, procedures, tests, and experiments.

3. Describe when provision 10 CFR 50.65(a)(4) of the Maintenance Rule should be used instead of 10 CFR 50.59.

4. Describe the applicable NRC regulation governing when a licensee may make changes to the fire protection program of a facility.

5. Evaluate example changes, tests, or experiments to determine whether the licensee may perform them without prior NRC approval and evaluate the example changes for their affect on operability.

6. Draft a notice of violation against 10 CFR 50.59.

TASKS:

1. Review the references listed above.

2. Complete the training on the NRC Web site covering 10 CFR 50.59.

3. Review at least three recently documented examples of violations of 10 CFR 50.59.

4. Meet with your supervisor and demonstrate your understanding of 10 CFR 50.59, including your ability to satisfy the above evaluation criteria.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item ISA-ENG-3

Engineering Individual Study Activity

TOPIC: (ISA-ENG-4) Basic Codes, Standards, and Regulatory Guides for Engineering Support

PURPOSE: The purpose of this activity is to provide you with very fundamental knowledge of the basic NRC codes, RGs, and associated industry standards commonly used by engineering inspectors. This activity will also acquaint you with the requirements (codes), guidelines (RGs), and accepted methodologies (industry standards) for licensees to use in accomplishing various safety-related activities. Finally, this activity will prepare you to determine an individual licensee's commitment to RGs and standards.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours

REFERENCES: See attached listings of general and discipline-related references.

EVALUATION CRITERIA: At the completion of this activity, you should be able to do the following:

1. State the general code sections commonly used by engineering inspectors and discuss the topics included in these sections.

2. Discuss the relationship between RGs and industry standards.

3. Identify the RGs and associated industry standards that address the quality assurance criteria in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

4. Discuss the topics included in the RGs and industry standards associated with your engineering discipline.

5. Discuss the application of these references to engineering inspection activities.

TASKS:

1. Read Appendix B to 10 CFR Part 50 and review a selected licensee's quality assurance manual. Review a sample of licensee implementing procedures (such as those associated with engineering inspections—design control and corrective action) in

accordance with an evaluation criterion to explain how a typical licensee meets the requirements.

2. Review the references in the attached list of general references as well as those listed for your specific discipline.

3. Locate the listing of RGs on the NRC external Web page.

4. Review a plant-specific UFSAR to identify the licensee's commitments to particular RGs and standards.

5. Discuss with experienced inspectors any questions you have concerning the topics of the references or their application to inspection activities.

6. Meet with your supervisor to demonstrate your familiarity with the applicable references and discuss the applications of these references to engineering inspection activities.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item ISA-ENG-4

REFERENCES FOR ISA-ENG-4

General

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants”

10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”

10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors”

10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants”

10 CFR 50.55a, “Codes and Standards”

10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”

RGs (Appendix B to 10 CFR Part 50)	American National Standards Institute (ANSI) Standards
1.28	ASME NQA-1
1.33	ANSI 18.1
1.37	ANSI N45.1
1.38	ANSI N45.2.2–1972
1.39	ANSI N45.2.3
1.30	ANSI N45.2.4
1.94	ANSI N45.2.5
1.116	ANSI N45.2.8
1.54	ANSI N101.4

ANSI/American Nuclear Society (ANS) 58.14, “Safety and Pressure Integrity Classification Criteria for LWR”

RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”

RG 1.29, “Seismic Design Classification”

RG 1.100, “Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants”

RG 1.155, “Station Blackout”

RG 1.186, “Guidance and Examples for Identifying 10 CFR 50.2 Design Bases” (ADAMS Accession No. ML003754825)

NEI 97-04, “Design Basis Program Guidelines,” Appendix B (ADAMS Accession No. ML003771698)

Electrical

RG 1.6, “Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems” (ADAMS Accession No. ML0037739924)

RG 1.30, “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)” (ANSI N45.2.4/Institute of Electrical and Electronics Engineers (IEEE) 336)

RG 1.32, “Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants” (IEEE 308)

RG 1.40, “Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants” (IEEE 334)

RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems”

RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems” (IEEE 279 and IEEE 379)

RG 1.63, “Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants” (IEEE 317)

RG 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants”

RG 1.75, “Physical Independence of Electric Systems” (IEEE 384)

RG 1.81, “Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants”

RG 1.89, “Qualification of Class 1E Equipment for Nuclear Power Plants” (IEEE 323)

RG 1.106, “Thermal Overload Protection for Electric Motors on Motor-Operated Valves”

RG 1.128, “Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants” (IEEE 484)

RG 1.129, “Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants” (IEEE 450)

RG 1.131, “Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants” (IEEE 383)

RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”

Instrumentation and Control

RG 1.11, “Instrument Lines Penetrating Primary Containment”

RG 1.12, “Instrumentation for Earthquakes”

RG 1.30, “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment” (ANSI N45.2.4/IEEE 336)

RG 1.32, “Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants” (IEEE 308)

RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems”

RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems” (IEEE 279 and IEEE 379)

RG 1.62, “Manual Initiation of Protective Actions”

RG 1.63, “Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants” (IEEE 317)

RG 1.75, “Physical Independence of Electric Systems” (IEEE 384)

RG 1.89, “Qualification of Class 1E Equipment for Nuclear Power Plants” (IEEE 323)

RG 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident”

RG 1.105, “Instrument Set Points” (Instrument Society of America (ISA) S67.04)

RG 1.151, “Instrument Sensing Lines” (ISA S67.02)

Electric Power Research Institute (EPRI) TR-102348, “Guideline on Licensing Digital Upgrades” (ADAMS Accession No. ML02080169)

IEEE 7-4.3.2-1993, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations”

IEEE 1050-1996, “Guide for Instrument and Control Equipment Grounding in Generating Stations”

IEEE 338-1987, “IEEE Standard Criteria for Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems”

Mechanical

RG 1.1, “NPSH for ECCS & Containment Heat Removal System Pumps (Safety Guide 1)”

RG 1.9, “Design, Qualification & Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants”

RG 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants”

RG 1.27, “Ultimate Heat Sink for Nuclear Power Plants”

RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident”

RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”

RG 1.85, “Materials Code Case Acceptability—ASME Section III, Division 1” (incorporated into RG 1.84)

RG 1.102, “Flood Protection for Nuclear Power Plants”

RG 1.116, “QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems”

RG 1.137, “Fuel Oil Systems for Standby Diesel Generators”

RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”

RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1“

RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”

American Society of Mechanical Engineers (ASME) OM CODE-(year), “Code for Operation and Maintenance of Nuclear Power Plants” (Section ISI, “Rules for Inservice Testing of Light-Water Reactors”; Subsection ISTA, “General Requirements”; ISTB, “IST of Pumps”; ISTC, “IST of Valves”; Appendix I, “IST of Pressure Relief Devices”; Appendix II, “IST of Check Valves”)

ASME OM-S/G-(year), “Standards and Guides for Operation and Maintenance of Nuclear Power Plants”

NUREG-1482, “Guidelines for Inservice Testing at Nuclear Plants”

Civil

ASME OM CODE-(year), “Code for Operation and Maintenance of Nuclear Power Plants” (Section ISI, “Rules for Inservice Testing of Light-Water Reactors,” Subsection ISTD, “Inservice Examination and Evaluation of Snubbers”)

American Welding Society (AWS) DI.1, “Structural Welding Code”

American Concrete Institute (ACI) 311, “Recommended Practice for Concrete Inspection”

ACI 318, “Building Code Requirements for Reinforced Concrete”

ACI 349.3, “Evaluation of Existing Nuclear Safety Related Concrete Structures”

ACI 214-77, “Recommended Practice for Evaluation of Strength Test Results of Concrete,” 1983

ACI 304R-89, “Guide for Measuring, Mixing, Transporting, and Placing Concrete”

ACI 309R-87, “Guide for Consolidation of Concrete”

ACI 347R-88, “Guide to Formwork for Concrete”

RG 1.12, Revision 2, “Nuclear Power Plant Instrumentation for Earthquakes”

RG 1.35, Revision 3, “Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,”

RG 1.35.1, Revision 0, “Determining Prestressing Forces for Inspection of Prestressed Concrete Containments”

RG 1.59, Revision 2, “Design-Basis Floods for Nuclear Power Plants”

RG 1.60, Revision 1, “Design Response Spectra for Seismic Design of Nuclear Power Plants”

RG 1.61, Revision 0, “Damping Values for Seismic Design of Nuclear Power Plants”

RG 1.76, Revision 0, “Design-Basis Tornado for Nuclear Power Plants”

RG 1.102, Revision 1, “Flood Protection for Nuclear Power Plants”

RG 1.117, Revision 1, “Tornado Design Classification”

RG 1.122, Revision 1, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components”

RG 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants”

RG 1.132, Revision 1, “Site Investigations for Foundations of Nuclear Power Plants”

RG 1.136, Revision 2, “Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the “Code for Concrete Reactor Vessels and Containments”)”

RG 1.138, Revision 0, “Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants”

RG 1.142, Revision 2, “Safety Related Concrete Structures for Nuclear Power Plants”

RG 1.165, Revision 0, “Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion”

RG 1.166, Revision 0, “Pre-Earthquake Planning and Immediate Nuclear Plant Operator Postearthquake Actions”

RG 1.167, Revision 0, “Restart of a Nuclear Power Plant Shut Down by a Seismic Event”

ANSI N14.6, “Special Lifting Devices for Shipping Containers Weighing 10000 Pounds or More”

ANSI N45.2.5, “Supplemental QA Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel”

ANSI/ANS 58.2, “Design Basis for Protection of LWR Power Plants Against the Effects of Postulated Pipe Rupture”

ASME Boiler and Pressure Vessel (B&PV) Code, Sections III, V, IX, and XI

Metallurgical/Welding

AWS DI.7, “Structural Welding Code”

RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”

RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components”

RG 1.44, “Control of the Use of Sensitized Stainless Steel”

RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel”

RG 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants”

RG 1.71, “Welder Qualification for Areas of Limited Accessibility”

RG 1.84, “Design and Fabrication Code Case Acceptability”

RG 1.85, “Materials Code Case Acceptability”

RG 1.147, “Inservice Inspection Code Case Acceptability”

RG 1.150, “Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations”

RG 1.178, “An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping”

EPRI, “PWR Steam Generator Examination Guidelines”

EPRI, “Steam Generator Integrity Assessment Guidelines”

ASME B&PV Code, Sections III, V, IX, and XI

Generic Letter 90-05, “Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping”

Industry Standards

Industry standards endorsed by the above RGs

ASME B&PV Code, Sections III, V, and VIII

Engineering Individual Study Activity

TOPIC: (ISA-ENG-5) Significance Determination Process—Reactor Inspection Findings for At-Power Situations

PURPOSE: The Significance Determination Process (SDP), as described in Appendix A to IMC 0609, aids NRC inspectors and staff in determining the safety significance of inspection findings, including the categorization of individual findings into one of four response bands, using risk insights when appropriate. The SDP determinations for inspection findings and the performance indicator information are combined for use in assessing licensee performance. The purpose of this activity is for you gain the requisite knowledge, understanding, and practical ability such that upon completion of this activity, you will be able to apply the SDP to reactor inspection findings for at-power situations to determine their safety significance.

COMPETENCY AREAS: INSPECTION
TECHNICAL AREA EXPERTISE
REGULATORY FRAMEWORK

LEVEL OF EFFORT: 16 hours

REFERENCES:

1. IMC 0609, "Significance Determination Process"

2. IMC 0609, Attachment 0609.01, "Significance and Enforcement Review Process"

3. IMC 0609, Attachment 0609.02, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)"

4. IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations"

5. IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening"

6. IMC 0612, Appendix E, "Examples of Minor Issues"

7. Reference Site Risk-Informed Inspection Notebook (http://nrr10.nrc.gov/adt/dssa/spsb/webpages/srapage/sdpnotebooks/sdpindex.html)

EVALUATION CRITERIA: At the completion of this activity, you should be able to do the following:

1. Explain the purpose, objectives, and applicability of the SDP process.

2. Describe and discuss the objectives of the Initiating Events (IE), Mitigating Systems (MS), and Barrier Integrity (BI) cornerstones.

3. Screen findings using the SDP Phase 1 Screening Worksheet for IE, MS, and BI Cornerstones in Appendix A to IMC 0609.

4. Define the safety significance and give examples of Green, White, Yellow, and Red findings.

5. Discuss your role during the Significance and Enforcement Review Process, as described in IMC 0609, Attachment 1.

6. Discuss the process for appealing NRC characterization of inspection findings (i.e., the SDP appeal process), as described in IMC 0609, Attachment 2.

TASKS:

1. Read the referenced section of IMC 0609, with particular focus on Appendix A.

2. Go to the Reactor Oversight Process Web site and review a sample of Green, White, Yellow, and Red findings in each of the three cornerstones (if samples of each level of safety significance are posted).

3. Read the case studies below and answer the following questions for each:

   a. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.
   b. If you determine that the issue is minor, then this scenario is completed.
   c. If you determine that the issue is more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.
   d. Compare your conclusion with those given in the actual findings and case studies.
   e. Discuss your results with your supervisor or a qualified inspector.

4. Whenever possible, attend a significance determination and enforcement review panel. Discuss the rationale for the outcome/resolution of the panel with a qualified inspector.

5. Meet with your supervisor or a qualified inspector to discuss any questions you may have as a result of this training activity.

DOCUMENTATION: Operations Technical Proficiency-Level Qualification Signature Card Item ISA-ENG-5

Scenario A

During the Unit 1 spring 1R16 refueling outage (RFO), plant staff identified that control rod drive mechanism nozzle XX was leaking. Workers repaired the nozzle weld and returned the unit to operation for another cycle. When the unit was shut down for RFO 1R17, visual examination of the reactor vessel head revealed repeat leakage of the nozzle. Based on the 1R16 RFO leakage, licensee staff performed an embedded flaw repair in accordance with Section XI of the ASME Code. However, the licensee staff recently concluded that this repair method is inadequate to prevent recurrence of the original primary water stress-corrosion cracking.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be more than a minor one, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Inspection Report 0500313/2003-008, ADAMS Accession No. ML040340732.)

5. Discuss your results with your supervisor or a qualified inspector.

Scenario B

On September 26, 2002, Unit 1 was at 99-percent reactor power, coasting down for the RFO scheduled to begin on October 5. At 5:41 a.m., the Unit 1 control room received a condenser off-gas alarm. At 12:43 p.m., the condenser off-gas 182 alarm actuated again and the No. 2 steam generator main steamline N-16 monitor went into alarm. At 10:24 p.m., the N-16 alarm cleared and the reading continued to trend downward.

On September 27, 2002, at 12:19 a.m., the condenser off-gas 182 alarm cleared. At 10:25 a.m., the N-16 alarm returned. At 10:40 a.m., the condenser off-gas 182 alarm came in, followed by the condenser off-gas 182 Hi alarm at 10:51 a.m. At 1:06 p.m., these alarms cleared. These alarms came in twice more on this day. At 7:54 p.m., the condenser off-gas 182 alarm came in, and at 10:32 p.m., the condenser off-gas 182 HiHi alarm was reached. The alarms cleared in less than an hour.

On September 28, 2002, at 1:40 a.m., the Unit 1 control room operators commenced power reduction in response to the 1-02 steam generator tube leak. At 3:12 a.m., the Unit 1 control room operators performed a planned trip of the Unit 1 reactor.

Through subsequent inspection and testing, the licensee determined the source of the leakage to be a stress-corrosion crack initiating from the outer diameter surface in the U-bend region of tube R41C71 of the No. 2 steam generator. The licensee also determined through pressure testing that the tube failed to exhibit structural and accident leakage integrity margins consistent with the plant design and licensing basis.

An NRC inspection team independently reviewed eddy current test data from the previous (1RF08) inspection in 2001 for the specific tube location where the leakage developed in September 2002. The team found that a clearly detectable indication was present at the leak location during the previous outage (1RF08) inspection in 2001. The indication did not meet the reporting criteria in the RFO 1RF08 analysis guidelines, and therefore neither the primary nor secondary analyst reported it in 2001.

The inspection team concluded that an experienced analyst should have recognize that the large wobble signal could mask a dent that could distort or rotate an indication outside the reportable phase angle response criteria. In such a case, the guidelines enabled the analyst to bring the indication to the attention of the lead analyst and the senior analyst. Therefore, the team determined that the analyst should have recognized the large wobble signal and should have brought it to the attention of a senior analyst.

As a direct consequence of the failure to detect the flaw, the tube was not removed from service and subsequently degraded to the point that it leaked and no longer satisfied the applicable tube integrity performance criteria. This occurred because the examination methods and the analysis guidelines used during the RFO were not effective for ensuring that tubes would maintain their integrity until the next scheduled inspection.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if

the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be a minor one, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Inspection Report 0500445, ADAMS Accession Nos. ML030090566, ML040270203, ML040440201, and ML040790025.)

5. Discuss your results with your supervisor or a qualified inspector.

Scenario C

The assumptions regarding the instruments used for safety-related heating, ventilation, and air-conditioning (HVAC) systems (i.e., the auxiliary building ventilation system and the control room HVAC system in the licensee's 120 volt alternating current (Vac) degraded voltage calculation) did not reflect the actual plant configuration. Specifically, the 120 Vac degraded voltage calculation, "Evaluation of the 120 Vac Distribution Circuits Voltage at the Degraded Voltage Setpoints," assumed the input voltage to specific HVAC process instrumentation to be at 95 Vac. While the vendor information associated with the instrumentation specified a higher voltage for proper operation, the licensee had stated in the assumption for the calculation that the instrumentation would be able to operate because tests on the instrumentation while in service demonstrated that the control circuits would perform their design function at a reduced voltage of 95 Vac. It was unclear whether the licensee had a program in place for testing replacement instrumentation put in service at this reduced voltage. Without a test for each instrument placed in service, the licensee would have to use the vendor's specification for voltage as it could not guarantee that the replacement instruments would operate at these assumed reduced voltages.

While the licensee was able to determine the operability of the affected instruments through the bounding voltage drop calculation, the licensee's existing design basis (the assumptions in the degraded voltage calculation) had not been adequately verified or maintained. The design-basis assumption relied on testing the instruments at 95 Vac; however, the licensee did not test some instruments and replaced others without retesting the specific instrument at the assumed degraded voltage included in the calculation. Therefore, the licensee had failed to maintain accurate design-basis assumptions that were essential for its design-basis calculation.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be minor, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Inspection Report 0500456/2003-007, ADAMS Accession No. ML032870193.)

5. Discuss your results with your supervisor or a qualified inspector.

Scenario D

The licensee did not identify potential common-mode failures that existed involving power supplies to the recirculation line air-operated valve in the auxiliary feedwater system and other system components. In addition, the licensee's corrective actions for the potential common-mode failure associated with a loss of instrument air did not prevent the failures from repeating. Although the licensee upgraded the safety function of the air-operated recirculation valve, this corrective action failed to ensure that successful operation of the recirculation line air-operated valve depended only on safety-related support systems. After the corrective actions, successful operation of the valve still depended upon nonsafety-related power to an interposing relay. In addition, the corrective actions did not to discover a single failure mechanism involving a system orifice modification.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be minor, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Inspection Report 0500266/2002-015, ADAMS Accession No. ML030920128.)

5. Discuss your results with your supervisor or a qualified inspector.

Scenario E

During an RFO, the licensee tested a charging pump at full-flow conditions as required every 18 months. Vibration data taken during this test indicated vibration of 0.324 inches per second (ips), which exceeded the test procedure alert range of 0.320 ips. The procedure required the surveillance frequency to be increased to every 9 months after exceeding the alert range. The licensee failed to identify that the test result exceeded the alert range and did not increase the test frequency. Subsequent vibration testing revealed no further vibration degradation. The ASME Code acceptance criterion for vibration measurements is 0.325 ips.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be a minor one, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Appendix E to IMC 0612.)

5. Discuss your results with your supervisor or a qualified inspector.

Scenario F

The licensee failed to consider one maintenance preventable functional failure (MPFF) of a system component during its demonstration of the effectiveness of preventive maintenance, in accordance with the Maintenance Rule (10 CFR 50.65(a)(2)). The Maintenance Rule requires, in part, that monitoring as specified in 10 CFR 50.65(a)(1) is not required if the licensee can demonstrate that it is effectively controlling the performance or condition of a structure, system, or component (SSC) through appropriate preventive maintenance, such that the item remains capable of performing its intended function. When the additional MPFF was considered, the conclusion from the demonstration remained valid.

Based on this scenario, complete the following steps:

1. Using Appendices B and E to IMC 0612, determine if the issue is more than a minor one. List the key conditions of the scenario that you will consider in determining if the issue is more than a minor one and that could be used to determine the safety significance of the issue.

2. If you determine the issue to be a minor one, proceed to step 5.

3. If you determine the issue to be more than a minor one, use the SDP Phase 1 Screening Worksheet in Appendix A to IMC 0609 to determine if the issue is of Green significance or if more analysis is required. Do not perform the additional analysis—do not perform a phase II or phase III SDP. Be able to justify your determination.

4. Compare your conclusions with those given in the actual findings or case studies. (See Appendix E to IMC 0612.)

5. Discuss your results with your supervisor or a qualified inspector.

Engineering On-the-Job Activity

Engineering On-the-Job Activity

TOPIC: (OJT-ENG-1) Safety System Design and Performance Capability

PURPOSE: The purpose of this activity is to do the following:

1. Familiarize you with activities commonly performed by an inspector while participating as a member of an SSDI team.

2. Observe and perform portions of the SSDI, as assigned by the team leader, using IPs 71111.21 and 71152.

3. Provide you with the opportunity to locate and identify the design- and licensing-basis requirements for a safety system and determine if those requirements are met and maintained.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours in-office preparation
80 hours onsite inspection

REFERENCES:

1. IP 71111.21, "Safety System Design and Performance Capability"

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

3. IMC 1245, On-the-Job Activity 4, "Inspection"

4. IMC 0612, "Power Reactor Inspection Reports"

5. Site-specific inspection plan (provided by team leader)

6. Site-specific design-basis documents (e.g., system descriptions, calculations, accident analyses, etc.)

7. Site-specific licensing basis (e.g., UFSAR, technical specifications, license amendments, and license amendment requests)

8. Licensee-provided preparation information (e.g., lists for applicable calculations, equipment history, problem reports, engineering evaluations, modifications, and procedures)

EVALUATION CRITERIA:

1. Complete the activities as outlined in this guide and meet with your supervisor to discuss any questions you may have as a result of completing this activity. Upon completion of the tasks in

this guide, you will be asked to demonstrate your understanding of the baseline IP 71111.21.

2. Demonstrate your ability to conduct inspection activities as applied to an SSDI (IP 71111.21).

3. Demonstrate your ability to locate and identify design- and licensing-basis information.

4. Demonstrate your familiarity with the design and licensing bases for the system(s) selected by the SSDI team inspection plan. Identify critical parameters and performance criteria.

5. Demonstrate your ability to identify critical equipment required to achieve the design-basis function of the selected system(s).

6. Demonstrate your ability to develop an individualized inspection plan for the discipline/system/equipment you are assigned from the team inspection plan.

7. Discuss your conclusions regarding the capability of your assigned equipment/system(s) to achieve its design- and licensing-basis functions. Provide the bases for that conclusion (e.g., evaluations, testing, performance history, etc.).

8. Demonstrate your capability to document your inspection findings consistent with IMC 0612.

9. Demonstrate your familiarity with SDP Group 1, 2, and 3 questions in IMC 0612 for an actual or simulated finding.

TASKS:

1. Perform the tasks listed in Entry-Level On-the-Job Activity 4, “Inspection Activities,” as applied to an inspection focused on IP 71111.21.

2. Review IP 71111.21 and IP 71152 for an overview of SSDI activity.

3. Review previous SSDI reports to improve your understanding of the implementation of IP 71111.21.

4. Review site-specific design- and licensing-basis documentation, provided during preparation week, to become familiar with the design and licensing bases for the systems selected for review in the team inspection plan.

5. Develop an individualized inspection plan for the system(s)/equipment you are assigned.

6. Identify specific critical equipment required for the safety system to achieve its design- and licensing-basis functions.

7. Review available information to determine if equipment is capable of achieving and maintaining its design function. Such information includes vendor manuals, specification documents, maintenance and testing documents, problem identification reports, etc.

8. Based on your inspection activity, assess if the system/equipment is capable of meeting its design function.

9. Perform a walkdown of accessible portions of the selected systems and equipment.

10. For at least one observed or simulated finding, apply SDP to the issue.

11. Meet with your supervisor or a qualified inspector designated by your supervisor and discuss the result of your activities.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item OJT-ENG-1

Engineering On-the-Job Activity

TOPIC: (OJT-ENG-2) Permanent Plant Modifications

PURPOSE: The purpose of this activity is to do the following:

1. Familiarize you with activities commonly performed by an inspector while reviewing permanent plant modifications.

2. Observe and perform portions of an inspection of permanent plant modifications using IP 71111.17.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours

REFERENCES:

1. IP 71111.17, “Permanent Plant Modifications”

2. Criterion III, “Design Control,” in Appendix B to 10 CFR Part 50

3. IMC 1245, Entry-Level On-the-Job Activity 4, “Inspection Activities”

4. IMC 0612, “Power Reactor Inspection Reports”

5. ANSI Standard N45.2.11-1974, “Quality Assurance Requirements for the Design of Nuclear Power Plants”

EVALUATION CRITERIA: Complete the activities as outlined in this guide and meet with your supervisor to discuss any questions you may have as a result of completing this activity. Upon completion of the tasks in this guide, you will be asked to demonstrate your understanding of how to conduct an inspection of plant modifications using the baseline IP 71111.17 by doing the following:

1. Demonstrate your ability to satisfy the evaluation criteria given in Entry-Level On-the-Job Activity 4, as applied to an inspection focused on IP 71111.17.

2. Demonstrate your ability to select modifications for review that are risk significant.

3. For selected modifications, demonstrate your ability to identify the design safety function of the SSC and the design requirements.

4. For each modification, discuss the licensee's approach to assuring that the modification did not adversely impact the design, availability, reliability, or functional capability of the SSC.

5. Demonstrate an understanding of potential risk-significant plant configurations that could occur during modification implementation and identify the licensee's method for addressing them.

6. Demonstrate your ability to document your inspection findings consistent with IMC 0612.

7. Demonstrate your familiarity with the SDP Group 1, 2, and 3 questions in IMC 0612 for an actual or simulated finding.

TASKS:

1. Perform the tasks listed in Entry-Level On-the-Job Activity 4, as applied to an inspection focused on IP 71111.17.

2. With the regional probabilistic risk assessment specialist, discuss which systems or equipment modifications have the highest risk significance.

3. For the modifications selected, determine the intended safety function and design requirements for the applicable SSC.

4. For the modifications selected, determine the motivation for the change (e.g., problem report, equipment failure, etc.) and verify that the modification resolved the problem.

5. Review testing and inspection documentation after the modification and verify that the testing was adequate to assure that the functional capability or design function of the SSC was not degraded.

6. Review the plant configuration for modification implementation and testing. Review the licensee's actions to assure that the plant was not placed in a risk-significant configuration.

7. When possible, perform a field walkdown of the SSC modified and determine whether the final condition was as designed by the modification documentation.

8. For a change in or substitution of component parts via the procurement or modification process, review equivalency evaluations that validate the adequacy of the replacement part.

9. For at least one observed or simulated finding, apply SDP to the issue.

10. Meet with your supervisor or a qualified inspector designated by your supervisor and discuss the result of your activities.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item OJT-ENG-2

Engineering On-the-Job Activity

TOPIC: (OJT-ENG-3) Inspection of Licensee Changes, Tests, and Experiments (10 CFR 50.59)

PURPOSE: The purpose of this activity is to do the following:

1. Familiarize you with activities commonly performed by an inspector while inspecting licensee changes, tests, and experiments to determine if they may be accomplished before receiving NRC approval.

2. Observe and perform portions of an inspection of changes, tests, and experiments using IP 71111.02.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours

REFERENCES:

1. IP 71111.02, "Evaluations of Changes, Tests, or Experiments"

2. Engineering Individual Study Activity ISA-ENG-3 on 10 CFR 50.59

3. Entry-Level On-the-Job Activity 4, "Inspection Activities"

EVALUATION CRITERIA: Complete the activities as outlined in this guide and meet with your supervisor to discuss any questions you may have as a result of completing this activity. Upon completion of the tasks in this guide, you will be asked to demonstrate your understanding of the baseline IP 71111.02 by doing the following:

1. Demonstrate your ability to satisfy the evaluation criteria of Entry-Level On-the-Job Activity 4, as applied to an inspection focused on IP 71111.02.

2. Describe the changes, tests, or experiments that you reviewed and your evaluation of the licensee's ability to perform them without prior NRC approval and their effect, if any, on operability.

TASKS:

1. Perform the tasks listed in Entry-Level On-the-Job Activity 4, as applied to an inspection focused on the topic of 10 CFR 50.59 (IP 71111.02).

2. For at least one observed or simulated finding, relating to 10 CFR 50.59, apply SDP.

3. Meet with your supervisor or a qualified inspector designated by your supervisor and discuss the result of your activities.

DOCUMENTATION: Engineering Proficiency-Level Qualification Signature Card Item OJT-ENG-3

Engineering On-the-Job Activity

TOPIC: (OJT-ENG-4) Security Plan and Implementation

PURPOSE: The purpose of this activity is to familiarize you with a typical security plan for a nuclear facility.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 12 hours

REFERENCES:

1. Security plan for a selected facility
2. Technical specifications for the selected facility
3. 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Radiological Sabotage"

EVALUATION CRITERIA: Upon completion of the tasks, you should be able to perform the following:

1. Generally describe the methods used by the site security force to maintain access control of the owner-controlled, protected, and vital areas.
2. Demonstrate the appropriate procedures for escorting visitors into and out of the protected and vital areas.
3. Explain the site-specific methods used to detect intruders.
4. Explain the need to maintain classification of certain safeguards material.

TASKS:

1. Review the references listed above, as appropriate, to develop an understanding of the site security system.
2. Conduct a walkdown of the protected and vital areas to identify the various types of intruder-detection equipment used.
3. Tour the central and secondary alarm stations. Discuss the duties and responsibilities of personnel stationed in those facilities with the watchstanders and the security shift supervisor.

4. Discuss inspector responsibilities related to site security and safeguards with your supervisor or a qualified operations or physical security inspector. Include practical circumstances that you may encounter, such as the loss of a security badge or the identification of an inattentive guard. In addition, discuss any questions that you may have as a result of this activity.

5. Meet with your supervisor or a qualified inspector designated by your supervisor and discuss the result of your activities.

DOCUMENTATION: Engineering Inspection Proficiency-Level Qualification Signature Card Item OJT-ENG-4

Engineering Inspector On-the-Job Activity

TOPIC: (OJT-ENG-5) Radiation Protection Program and Implementation

PURPOSE: The radiation protection program and implementing procedures are intended to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine nuclear reactor operation. Licensee procedures, 10 CFR Part 19, "Notes, Instructions and Reports to Workers: Inspection and Investigations," and 10 CFR Part 20, "Standards for Protection Against Radiation," address the as-low-as-reasonably-achievable (ALARA) program, external exposure, internal exposure, respiratory protection, posting and labeling, survey, and reporting requirements. This activity will provide you with a general understanding of the applicable regulatory requirements, the licensee's radiation protection program, and implementing procedures.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 16 hours

REFERENCES:

1. Licensee procedures addressing the implementation of NRC inspections of the radiation protection program

2. Plant technical specifications

3. Plant UFSAR

4. 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations"

5. 10 CFR Part 20, "Standards for Protection Against Radiation"

6. RG 8.38, "Control of Access to High and Very High Radiation Areas"

7. Radiation work permit used for NRC inspection activities

EVALUATION CRITERIA: Upon completion of the tasks, you should be able to perform the following:

1. Generally describe the following terms and provide examples of each term:
   a. controlled area
   b. radiological restricted area

c. radiation area
d. high radiation area
e. locked high radiation area
f. very high radiation area
g. hot spots
h. airborne radiation area

2. Explain the ALARA concept and its application to the performance of radiological work at your site.

3. Describe the plant's overall administrative procedures for control of external, internal, and airborne exposure and its process for implementing the procedures during NRC inspections.

4. Describe physical and administrative controls for radiation areas, high radiation areas, very high radiation areas, and airborne radioactivity areas.

TASKS:

1. Locate the listed references for a selected facility.

2. Review the references and licensee procedures to develop an overall understanding of the regulatory requirements and the implementation of the radiation protection program. Review the radiation work permit, which allows a visiting NRC inspector to complete the assigned inspection.

3. During a plant tour, identify at least one of each of the following—radiation area, high radiation area, very high radiation area, hot spots area, and an airborne radioactivity area. Observe the licensee's method of controlling access to each in accordance with regulations and licensee requirements.

4. Review at least one completed set of radiation survey results and explain how you will incorporate the survey results into your inspection effort.

5. Review the licensee procedures for radiation control. Review the actions required of an individual when contamination is detected before exiting the radiation controlled area.

6. Meet with your supervisor or a qualified engineering inspector to discuss any questions that you may have as a result of these activities and demonstrate that you can meet the evaluation criteria.

DOCUMENTATION: Engineering Inspector Proficiency-Level Qualification Signature Card Item OJT-ENG-5

Reactor Engineering Technical Proficiency-Level Signature Card and Certification

Inspector Name: ______________________________	*Employee Initials/Date*	*Supervisor's Signature/Date*
A. Training Courses		
Power Plant Engineering (self study)		
Reactor Full Series (either BWR or PWR)		
Basic Reactor Operations for alternate reactor type		
B. Individual Study Activities		
ISA-ENG-1 Capabilities of Safety Systems to Perform Intended Safety Functions		
ISA-ENG-2 The NRC's Review of Temporary and Permanent Plant Modifications		
ISA-ENG-3 Evaluations of Changes, Tests, and Experiments (10 CFR 50.59)		
ISA-ENG-4 Basic Codes, Standards, and Regulatory Guides for Engineering Support		
ISA-ENG-5 Significance Determination Process—Reactor Inspection Findings for At-Power Situations		
C. On-the-Job Training Activities		
OJT-ENG-1 Safety System Design and Performance Capability		
OJT-ENG-2 Permanent Plant Modifications		
OJT-ENG-3 Inspection of Licensee Changes, Tests, and Experiments (10 CFR 50.59)		
OJT-ENG-4 Security Plan and Implementation		
OJT-ENG-5 Radiation Protection Program and Implementation		

Supervisor's signature indicates successful completion of all required courses and activities listed in this journal and readiness to appear before the Oral Board.

Supervisor's Signature: ______________________________ Date: _______________

The appropriate Form 1, "Reactor Engineering Technical Proficiency-Level Equivalency Justification," if applicable, must accompany this signature card and certification.

Copies: Inspector
Human Resources Office
Supervisor

Form 1: Reactor Engineering Technical Proficiency-Level Equivalency Justification	
Inspector Name: ______________________________	*Identify equivalent training and experience for which the inspector is to be given credit*
A. Training Courses	
Power Plant Engineering (self study)	
Reactor Full Series (either BWR or PWR)	
Basic Reactor Operations for alternate reactor type	
B. Individual Study Activities	
ISA-ENG-1 Capabilities of Safety Systems to Perform Intended Safety Functions	
ISA-ENG-2 The NRC's Review of Temporary and Permanent Plant Modifications	
ISA-ENG-3 Evaluations of Changes, Tests, and Experiments (10 CFR 50.59)	
ISA-ENG-4 Basic Codes, Standards, and Regulatory Guides for Engineering Support	
ISA-ENG-5 Significance Determination Process—Reactor Inspection Findings for At-Power Situations	

C. On-the-Job Training Activities	*Identify equivalent training and experience for which the inspector is to be given credit*
OJT-ENG-1 Safety System Design and Performance Capability	
OJT-ENG-2 Permanent Plant Modifications	
OJT-ENG-3 Inspection of Licensee Changes, Tests, and Experiments (10 CFR 50.59)	
OJT-ENG-4 Security Plan and Implementation	
OJT-ENG-5 Radiation Protection Program and Implementation	

Supervisor's Recommendation

Signature/Date___________________________

Division Director's Approval Signature/Date___________________________

Copies to: Inspector
Human Resources Office
Supervisor

Revision History Sheet

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	10/31/06 CN 06-032	To update reference lists and incorporate minor editorial changes. Completed 4 year historical CN search	None	N/A	N/A