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

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Table of Contents

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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
(These courses have the completion of Appendix A as a prerequisite.)

- Reactor Full Series (either BWR or 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
  - If you completed the BWR series then you must take R-104P
  - If you completed the PWR series then you must take R-104B

(This 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
This course IS NOT required for initial qualification. Attendance at this course is a post-qualification requirement to be completed with 24 months of full qualifications.

- PRA Technology and Regulatory Perspectives (P-111)

Required Refresher Training

One of the following is required once every three years:

- BWR or PWR systems refresher OR
- Simulator refresher OR
- EOP refresher

If you completed your qualification before the requirement for a full series was added, you should alternate between R-104B and R-104P. Your selection should be coordinated with your supervisor.

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Engineering Individual Study Guide

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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 safety functions of those systems.

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 32 hours

REFERENCES:

1. 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 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Basis (ADAMS Accession Number: ML003754825)

6. NEI 97-04, "Design Basis Program Guidelines, Appendix B, (ADAMS Accession Number ML003771698)

7. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

8. Inspection Manual Chapter 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 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 which 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 basic and the basis for their importance.

2. Discuss what the purposes are for a Safety System Design Inspection (SSDI:
   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 a SSDI and the various methods for that selection process. The emphasis should be on some measure which 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 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 or not.

6. Define the contribution of each of the following documents to a SSDI, and what is its benefit in determining the functional capability of one or more systems.
   a. design basis documents
   b. licensing basis
   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 (IN, BL, and GL)

7. For the listed documents in previous question, state how each provides insights into the assessment of a licensee's quality assurance program at least in 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 with the minimum criteria that must be present to derive a similar reasonable affirmation.

9. Define which reactor-oversight program cornerstones are verified by the reviews of safety systems via a 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 Evaluation Criteria.

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

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Engineering Individual Study Guide

TOPIC: (ISA-ENG-2) 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 basis and licensing basis, and also 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 the monitoring of 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 why important.

2. Discuss the typical format of both permanent and temporary modifications (major sections and purpose of each).
3. Discuss how licensees control modifications both prior and post implementation including affected design documents and plant procedures.

4. Define the following terms:
   a. Configuration management

b. Current Licensing Bases
c. Design
d. Design Bases
e. Design-Bases 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 following items - 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 what is the reason for each one’s inclusion.

7. State at least five of the types of changes that comprise the category “temporary plant modifications” and what is the reason for each one’s inclusion.

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 which 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 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 (10CFR 50.59)

PURPOSE: The purpose of this activity is to acquaint you with how to review safety evaluations which are used to determine if the power reactor facility change, test, or experiment, requires NRC approval prior to 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 basis and licensing basis, and also the performance capability of safety systems and components. The purpose is to:

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

2. Enable you to demonstrate an ability to conduct a 50.59 inspection in accordance with inspection procedure (IP) 71111.02

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 24 hours

REFERENCES: 10 CFR 50.59; Changes, Tests, and Experiments

NRC Regulatory Guide 1.187; Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments

NEI 96-07, Revision 1; Guidelines for 10 CFR 50.59 Implementation

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

Regulatory Issue Summary (RIS) 2001-03; Changes, Tests, and Experiments

RIS 2001-09; Control of Hazard Barriers (guidance on 50.59 applicability to barriers).

Current Regional or Office guidance for processing potential violations of 10 CFR 50.59

IP 71111.02; Evaluations of Changes, Tests, or Experiments

IP 71152, Identification and Resolution of Problems

IP 71111.15, Operability Evaluations

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

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

2. State the meaning of key terms used in this regulation: UFSAR, changes, facility, procedures, tests or 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 for whether the licensee may perform them without prior NRC approval. Also, 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 NRC WEB site training on 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 Engineering Inspectors with a very fundamental knowledge with the basic NRC codes, regulatory guides (RGs) and associated industry standards commonly used by Engineering Inspectors. This activity will also provide you with a basic acquaintance with the requirements (codes), guidelines (RGs) and accepted methodologies (industry standards) for licensee's to accomplish various safety related activities. And lastly 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:

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 which address the 10 CFR 50 Appendix B, Quality Assurance Criteria.

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

10. Discuss application of these references to engineering inspection activity.

TASKS:

1. Read 10 CFR 50, Appendix B and review a selected licensee's Quality Assurance Manual. A sample of licensee implementing procedures should be reviewed (such as those associated with engineering inspections - Design Control

and Corrective Action) with an Evaluation Criteria 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 on the NRC external web page the listing of Regulatory Guides.

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

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

6. Meet with your supervisor and 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.

<u>REFERENCES for ISA-ENG-4</u>

General

10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants

10CFR 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

Regulatory Guides (10 CFR 50 Appendix B)	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/ANS 58.14 Safety and Pressure Integrity Classification Criteria for LWR

Regulatory Guide 1.100, Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants

Regulatory Guide 1.26 Quality Group Classifications and Standards for Water-,Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Regulatory Guide 1.29, Seismic Design Classification

Regulatory Guide 1.155, Station Blackout

Regulatory Guide 1.186, Guidance and Examples for Identifying 10 CFR 50.2 Design Bases (ADAMS Accession Number: ML003754825)

NEI 97-04, Design Basis Program Guidelines, Appendix B, (ADAMS Accession Number of Appendix B of NEI 96-04 : ML003771698)

Electrical

Regulatory Guide 1.6, Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (ADAMS ML0037739924

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

Regulatory Guide 1.32, Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (IEEE 308)

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

Regulatory Guide 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

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

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

Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Regulatory Guide 1.75, Physical Independence of Electric Systems (IEEE 384)

Regulatory Guide 1.81, Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants

Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants (IEEE 323)

Regulatory Guide 1.106, Thermal Overload Protection for Electric Motors on Motor-Operated Valves

Regulatory Guide 1.128, Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (IEEE 484)

Regulatory Guide 1.129, Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (IEEE 450)

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

Regulatory Guide 1.180, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems

<u>**Instrumentation & Control**</u>

Regulatory Guide 1.11, Instrument Lines Penetrating Primary Containment

Regulatory Guide 1.12, Instrumentation for Earthquakes

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

Regulatory Guide 1.32, Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (IEEE 308)

Regulatory Guide 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

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

Regulatory Guide 1.62, Manual Initiation of Protective Actions

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

Regulatory Guide 1.75, Physical Independence of Electric Systems (IEEE 384)

Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants (IEEE 323)

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

Regulatory Guide 1.105, Instrument Set points (ISA S67.04)

Regulatory Guide 1.151, Instrument Sensing Lines (ISA S67.02)

EPRI TR-102348	Guideline on Licensing Digital Upgrades (ADAMS 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

Regulatory Guide 1.1,	NPSH for ECCS & Containment Heat Removal System Pumps (safety guide 1)
Regulatory Guide 1.9,	Design, Qualification & Testing of EDGs used as Class 1E Onsite Electrical Power Systems
Regulatory Guide 1.26	Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants
Regulatory Guide 1.27,	Ultimate Heat Sink for Nuclear Power Plants
Regulatory Guides	1.84, 1.85, and 1.147 ASME Code Case Applicability.
Regulatory Guide 1.82	Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident
Regulatory Guide 1.102	Flood Protection for Nuclear Power Plants
Regulatory Guide 1.116,	QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems
Regulatory Guide 1.137,	Fuel Oil Systems for Standby Diesel Generators
Regulatory Guide 1.140,	Design, Inspection, & Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Clean Up Systems
Regulatory Guide 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
ASME OM CODE-(year),	Code for Operation and Maintenance of Nuclear Power Plants (Section ISI, Rules fro Inservice Testing of LWR) Subsections 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 LWR) Subsection, ISTD, Inservice Examination and Evaluation of Snubbers
AWS DI.1	Structural Welding Code
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
Regulatory Guide 1.12,	Nuclear Power Plant Instrumentation for Earthquakes, Rev. 2
Regulatory Guide 1.35,	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments, Rev. 3
Regulatory Guide 1.35.1,	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, Rev. 0
Regulatory Guide 1.59,	Design Basis Floods for Nuclear Power Plants, Rev. 2
Regulatory Guide 1.60,	Design Response Spectra for Seismic Design of Nuclear Power Plants, Rev. 1

Regulatory Guide 1.61,	Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0
Regulatory Guide 1.76,	Design Basis Tornado for Nuclear Power Plants, Rev. 0
Regulatory Guide 1.102,	Flood Protection for Nuclear Power Plants, Rev. 1
Regulatory Guide 1.117,	Tornado Design Classification, Rev. 1
Regulatory Guide 1.122,	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components, Rev. 1
Regulatory Guide 1.127,	Inspection of Water-Control Structures Associated with Nuclear Power Plants, Rev.
Regulatory Guide 1.132,	Site Investigations for Foundations of Nuclear Power Plants, Rev. 1
Regulatory Guide 1.136,	Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the Code for Concrete Reactor Vessels and Containments), Rev. 2
Regulatory Guide 1.138,	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants, Rev. 0
Regulatory Guide 1.142,	Safety Related Concrete Structures for Nuclear Power Plants, Rev. 2
Regulatory Guide1.165,	Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion, Rev. 0
Regulatory Guide1.166,	Pre-Earthquake Planning and Immediate Nuclear Plant Operator Postearthquake Actions, Rev. 0
Regulatory Guide1.167,	Restart of a Nuclear Power Plant Shut Down by a Seismic Event, Rev. 0
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
ASME B&PV	Section III, V, IX, XI

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

Metallurgical / Welding
AWS DI.7 Structural Welding Code

Regulatory Guide 1.31	Control of Ferrite Content in Stainless Steel Weld Metal
Regulatory Guide 1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components
Regulatory Guide 1.44	Control of the Use of Sensitized Stainless Steel
Regulatory Guide 1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel
Regulatory Guide1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations
Regulatory Guide1.71	Welder Qualification for Areas of Limited Accessibility
Regulatory Guide 1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants
Regulatory Guide 1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations
Regulatory Guide1.178	An Approach For Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping
Regulatory Guide 1.84	Design and Fabrication Code Case Acceptability
Regulatory Guide 1.85	Materials Code Case Acceptability
Regulatory Guide 1.147	Inservice Inspection Code Case Acceptability
EPRI	PWR Steam Generator Examination Guidelines
EPRI	Steam Generator Integrity Assessment Guidelines
ASME B&PV	Section III, V, IX, XI
GL90-05	Temporary Non-Code Repair of ASME Code Class 1,2,3 Piping

Industry Standards
Industry Standard endorsed by the above Regulatory Guides
ASME Boiler and Pressure Vessel Code, Sections III, V, and VIII

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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 of Manual Chapter 0609, aids NRC inspectors and staff in determining the safety significance of inspection findings, including 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 use the Significance Determination of Reactor Inspection Findings for At-Power Situations to determine the safety significance of reactor inspection findings.

COMPETENCY AREAS: INSPECTION
TECHNICAL AREA EXPERTISE
REGULATORY FRAMEWORK

LEVEL OF EFFORT: 16 hours

REFERENCES:

1. Inspection Manual Chapter (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 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:

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

2. Describe and discuss the objective of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones.

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

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 (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 ROP web-site and review a sample of Green, White, Yellow and Red findings in each of the three cornerstones (if there are samples of each safety siginificance).

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

    3. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

    4. If the issue is determined to be minor, then this scenario is completed.

    5. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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.

6. Compare your conclusion with those provided by the actual findings and case studies.

7. Discuss your results with your supervisor or a qualified Inspector.

4. Whenever possible, attend a significance determination and enforcement review panel (SERP). Discuss 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), Control Rod Drive Mechanism Nozzle XX was identified as leaking. Repairs were made to the nozzle weld, and the unit was returned to operation for another cycle. Upon shutdown for RFO 1R17, repeat leakage of the nozzle was self revealed during visual examination of the reactor vessel head. 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 was inadequate to prevent recurrence of the original primary water stress corrosion cracking.

4. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

5. If the issue is determined to be minor, then go to 5

6. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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.

7. Compare your conclusions with those provided by the actual findings or case studies. [IR 0500313/2003-008, ML040340732]

8. 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 with the refueling outage 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 steam line 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 and at 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.

Subsequent inspection and testing by 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 Steam Generator 2. 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 was outside (did not meet) the reporting criteria in the Refueling Outage 1RF08 analysis guidelines and was not reported by either the primary or secondary analyst 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. 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.

The direct consequence of failure to detect the flaw was that 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 used during RFO, including the analysis guidelines, were not effective for ensuring that tubes would maintain their integrity until the next scheduled inspection.

1. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

2. If the issue is determined to be minor, then go to 5

3. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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 provided by the actual findings or case studies. [IR 0500445: ML030090566, ML040270203, ML040440201, ML040790025]

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

Scenario C

The assumptions in regard to instruments used for safety related HVAC systems (the auxiliary building ventilation system and the control room HVAC System in the licensee's 120 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 since tests on the instrumentation 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 vendor's specification for voltage would have to be used as it could not be guaranteed that the replacement instruments would operate at these assumed reduced voltages.

While the licensee was able to determine operability of the affected instruments through 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 upon testing of the instruments at 95 Vac; however, some instruments were either not tested, while others were replaced without retesting the specific instrument at the assumed degraded voltage included in the calculation. As such, the licensee had failed to maintain accurate design basis assumptions that were essential for their design basis calculation.

1. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

2. If the issue is determined to be minor, then go to 5

3. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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 provided by the actual findings or case studies. [IR 0500456/2003-007/ML032870193]

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

Scenario D

The licensee failed to identify potential common mode failures that existed involving power supplies to the recirculation line air-operated valve in the auxiliary feedwater (AFW) 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 preclude repetition. Specifically, the licensee's corrective actions, to upgrade the safety function of the air-operated recirculation valve, failed to ensure that successful operation of the recirculation line air-operated valve was dependent only on safety-related support systems. Following the corrective actions, successful operation of the valve was still dependent upon nonsafety-related power to an interposing relay. Additionally, the corrective actions failed to discover a single failure mechanism involving a system orifice modification.

1. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

2. If the issue is determined to be minor, then go to 5

3. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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 provided by the actual findings or case studies. [IR 0500266/2002-015, MLML030920128]

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

Scenario E

During a refueling outage, 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 nine months after exceeding the Alert range. The licensee failed to identify that the test result exceeded the Alert range, so the test frequency was not increased. Subsequent vibration testing revealed no further vibration degradation. The ASME Code acceptance criterion for vibration measurements was 0.325 ips.

1. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

2. If the issue is determined to be minor, then go to 5

3. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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 provided by the actual findings or case studies. [IMC 0612 Appendix E]

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 their a(2) demonstration that preventive maintenance was being effective. The Maintenance Rule 10 CFR 50.65(a)(2) requires, in part, that monitoring as specified in (a)(1) is not required where it has been demonstrated that the performance of condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. When the additional MPFF was considered, the a(2) conclusion remained valid.

1. Utilizing IMC-0612 Appendix B and Appendix E, determine if the issue is more than minor. List key conditions of the scenario that will be considered in determining if the issue is more than minor and that could be used to determine the safety significance.

2. If the issue is determined to be minor, then go to 5

3. If the issue is determined to be more than minor, use the SDP Phase 1 Screening Worksheet in IMC 0609 Appendix A to determine if the issues is Green 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 provided by the actual findings or case studies. [IMC 0612 Appendix E]

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

Engineering On-the-Job Activity

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Engineering On-the-Job Activity

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

PURPOSE: The purpose of this activity is to:

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

2. Observe and perform portions of the SSDI, as assigned by team leader, using Inspection Procedures (IPs) 71111.21 and 71152 (Identification and Resolution of Problems)

3. Provide you the opportunity to locate and identify the design and licensing base 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. MC 1245, On-the-Job Activity 4, Inspection

4. MC 0612, Power Reactor Inspection Reports

5. Site Specific Inspection Plan (provided by team leader)

6. Site Specific Design Basis Documents (DBDs, system descriptions, calculations, accident analyses, etc.)

7. Site Specific Licensing Basis (UFSAR, TS, Licence Amendments, and LARs)

8. Licensee provided preparation information, i.e. 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 inspection procedure 71111.21.

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

3. Demonstrate your ability to locate and identify the design base and licensing base 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 for 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 their design and licensing base functions. Provide the bases for that conclusion, i.e. evaluations, testing, performance history, etc.

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

9. Demonstrate your familiarity with the Significance Determination Process (SDP), Group 1, 2, and 3 questions of MC 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 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 bases 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 bases functions.

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

8. Make an assessment based on your inspection activity 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, process the issue through the significance determination process (SDP).

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.

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Engineering On-the-Job Activity

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

PURPOSE: The purpose of this activity is to:

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

2. Observe and perform portions of an inspection of permanent plant modifications using Inspection Procedure (IP) 71111.17

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours

REFERENCES:

1. NRC IP 71111.17, Permanent Plant Modifications

2. 10 CFR 50 Appendix B, Criterion III, Design Control

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

4. MC 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 inspection procedure 71111.17.

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

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

3. For selected modifications demonstrate your ability to identify the design safety function of the structure, system, or component (SSC) and the design requirements

4. Discuss for each modification, how the licensee assured 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. How does the licensee address this?

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

7. Demonstrate your familiarity with the Significance Determination Process (SDP), Group 1, 2, and 3 questions of 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 71111.17

2. Discuss with Regional PRA specialist, which systems or equipment modifications have the highest risk significance.

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

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

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

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

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

8. For change 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, process the issue through the significance determination process (SDP).

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.

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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:

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

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

COMPETENCY AREA: INSPECTION

LEVEL OF EFFORT: 40 hours

REFERENCES:

1. NRC Inspection Procedure (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 inspection procedure 71111.02 by:

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

2. Describing the changes, tests, or experiments that you reviewed and your evaluation of why the licensee may (or may not) perform them without prior NRC approval and why they did not (or did) affect operability.

TASKS:

1. Perform the tasks listed in Entry Level On-the-Job Activity 4, Inspection Activities, 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, process the finding through the significance determination process (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 Part 73.55

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

1. Generally describe how the site security force maintains 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 for maintaining 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. Your discussion should include practical circumstances that you may encounter such as loss of security badge or 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. As Low As Is Reasonably Achievable (ALARA) program, external exposure, internal exposure, respiratory protection, posting and labeling, survey, and reporting requirements are addressed in licensee's procedures and in 10 CFR Parts 19 and 20. This activity will provide you 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 in the Radiation Protection program.

2. Plant Technical Specifications, Plant Updated Final Safety Analysis Report, and 10 CFR Parts 19 and 20.

3. Regulatory Guide 8.38, Control of Access to High and Very High Radiation Areas.

4. Radiation Work Permit (RWP) used for NRC inspection activities.

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

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 how it is applied to performance of radiological work at your site.

2. Describe the plant's overall administrative procedures for control of external exposure, internal exposure, and airborne exposure and how the procedures are implemented during NRC inspections.

3. 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's procedures to develop an overall understanding of the regulatory requirements and how the radiation protection program is being implemented. Review the Radiation Work Permit (RWP) which allows a visiting NRC inspector to complete their 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 and observe how each access is controlled in accordance with regulations and the licensee's requirements.

4. Review at least one completed 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 prior to exiting the Radiation Controlled Area (FCA).

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 Functions		
ISA-ENG-2	NRC's Review of Temporary and Permanent Plant Modifications		
ISA-ENG-3	10 CFR 50.59: Changes, Tests and Experiments		
ISA-ENG-4	Codes and Standards 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		
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: ________________

This signature card and certification must be accompanied by the appropriate Form 1, Reactor Engineering Technical Proficiency Level Equivalency Justification, if applicable.

Copies: Inspector
HR Office
Supervisor

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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 Functions	
ISA-ENG-2	NRC's Review of Temporary and Permanent Plant Modifications	
ISA-ENG-3	10 CFR 50.59: Changes, Tests and Experiments	
ISA-ENG-4	Codes and Standards 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	
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
HR Office
Supervisor