

NRC INSPECTION MANUAL

IRIB

INSPECTION PROCEDURE 50001

STEAM GENERATOR REPLACEMENT INSPECTION

Effective Date: February 4, 2025

PROGRAM APPLICABILITY: 2515 C

50001-01 INSPECTION OBJECTIVES

- 01.01 To verify that engineering evaluations and design changes associated with steam generator (SG) replacement are completed in conformance with requirements in the facility license, the applicable codes and standards, licensing commitments, and the regulations.
- 01.02 To verify that SG removal and replacement activities maintain adequate nuclear and radiological safety.
- 01.03 To verify that the SG post-installation test program is technically adequate, in conformance with requirements, and satisfactorily implemented.
- 01.04 To verify appropriate shutdown risk assessments, work controls, and risk management actions are implemented during the SG removal and replacement activities.

50001-02 INSPECTION REQUIREMENTS

02.01 Inspection Scoping

Develop a site-specific inspection plan to select and review the safety-related aspects associated with the major phases of the SG replacement project (SGRP).

During inspection preparation, review the processes for SGRP and plans for identifying, tracking, and resolving problems related to SGRP to ascertain that they are consistent with plant processes.

This procedure lists in Appendix A, "Applicable Inspection Procedures," certain baseline inspection procedures (IPs) to be used to perform parts of the inspection effort.

02.02 Design and Planning Inspections

Conduct SG design and planning inspections in accordance with the inspection plan by performing selective inspections, consistent with the safety significance and inspection resources, of the following areas:

- a. Conduct SG replacement engineering and technical support inspections in accordance with the inspection plan by performing selective inspections that will:

1. Verify that selected design changes and modifications to systems, structures, and components described in the **Updated** Final Safety Analysis Report are reviewed in accordance with **Title 10 of the Code of Federal Regulations (10 CFR) Section 50.59. "Changes, tests and experiments."**
 2. Review key design aspects and modifications for the replacement SGs and other modifications associated with SG replacement. Where applicable, include design reviews for the provision of a temporary containment opening.
- b. Review the applicable engineering design, modification, testing, and **analyses** associated with SG lifting and rigging including: -(1) crane and rigging equipment, (2) SG component drop analysis, (3) safe load paths, (4) load lay-down areas, and (5) heavy load haul path. The inspection should focus on the impact of load handling activities on the reactor core, spent fuel and its cooling, and other plant support systems for the reactor unit and common systems for the other operating unit(s) at the site.
 - c. Review radiation protection program controls, planning, and preparation in the following areas:
 1. as low as reasonably achievable (ALARA) planning.
 2. dose estimates and dose tracking.
 3. exposure controls including temporary shielding.
 4. contamination controls.
 5. radioactive material management.
 6. radiological work plans and controls.
 7. airborne radioactivity effluent controls.
 8. radioactive waste and material handling, storage, and transportation.
 - d. Review SGRP activities in the following areas:
 1. Security considerations associated with vital and protected area barriers that may be affected during replacement activities.
 2. Where applicable at multi-unit sites, the controls and plans to minimize any adverse impact on the operating unit(s) and common systems.

02.03 Steam Generator Removal and Replacement Inspections

Conduct SG removal and replacement inspections in accordance with the inspection plan by performing selective inspections, consistent with the safety significance and inspection resources, of the following areas:

- a. Review the following welding and nondestructive examination (NDE) activities:
 1. Where applicable, special procedures for welding and NDE
 2. Training and qualifications for personnel performing welding and NDE
 3. NDE including radiography results and work packages for selected welds
 4. Completion of preservice NDE requirements for welds, where required
 5. Completion of baseline eddy current examination of new SG tubes, if applicable

- b. Review activities associated with lifting and rigging: preparations and procedures for rigging and heavy lifting including any required crane and rigging inspections, testing, equipment modifications, lay-down area preparations, and training of personnel.
- c. Observe portions of old and new SG cutting, movement, and reconnection inside and outside containment including cutting pipe connections, supports, installation of the temporary restraints, lifting, lowering, position changes, transfer, and transportation.
- d. Inspect old SG hold down bolts for degradation (if logistically feasible and with minimal resources) and any major structural modifications that are performed to facilitate SG replacement.
- e. Inspect activities associated with containment access and, where applicable, creation and restoration of temporary containment opening and containment leakage testing.
- f. Inspect the following activities throughout the process as appropriate:
 - 1. Establishment of operating conditions including defueling, RCS draindown, and system isolation and safety tagging/blocking
 - 2. Implementation of radiation protection controls
 - 3. Inspect controls for excluding foreign materials in the primary and secondary side of the SGs and in the related RCS openings
 - 4. Installation, use, and removal of temporary services directly related to the activities identified in this procedure
- g. Review radiological safety plans for temporary storage or disposal of old SGs and components

02.04 Post-installation Verification and Testing Inspections

Conduct SG post-installation verification and testing inspections in accordance with the inspection plan. Perform selective inspections, consistent with the safety significance and inspection resources, of the following areas:

- containment integrity and leakage testing, as applicable
- the licensee's post-installation inspections and verifications program and its implementation
- the conduct of RCS leakage testing and review the test results
- the conduct of the SG secondary side leakage testing and review the test results
- calibration and testing of instrumentation for both the primary (RCS) and secondary side (feedwater and main steam) systems affected by SG replacement
- the procedures for equipment performance testing required to confirm the design and to establish baseline measurements and the conduct of testing, to include post installation and power ascension

General Guidance

A SG replacement at a nuclear power plant is a significant modification and maintenance activity involving many different licensee disciplines and has historically required extensive contractor support. This activity can potentially affect the power plant safety analysis, the containment structure, and plant operational characteristics.

Comprehensive Nuclear Regulatory Commission (NRC) inspection of a SG replacement activities involves coordination of inspections to review engineering, safety assessment, operations, maintenance, welding, NDE, instrumentation, quality assurance, radiation protection, security, and testing. It is expected that inspections will be performed primarily by regional specialists and the resident inspectors.

Although early SG replacements were processed as license amendments, since 1989 all replacements have been accomplished under 10 CFR 50.59, that is, without the need for a license amendment (other than for technical specification changes). However, because of the engineering scope of a replacement project and its safety significance, licensees have voluntarily participated in extensive interactions with the regions and NRR technical branches. This interaction starts early (three or more years before scheduled replacement) in the design effort and prior to contracting a vendor for SG fabrication. It is expected that these interactions will continue with future SGRPs. Because of the complexity of the SGRP, technical support from NRR for engineering reviews during the inspections should be considered. Those engineering reviews should be coordinated through the NRR project manager.

All inspections of SG replacement activities should be performed in accordance with this procedure and the site-specific inspection plan developed to support this procedure. The site-specific inspection plan should be developed and administered by the regional office to integrate the inspection activities of the specialist inspectors.

The site-specific inspection plan should include the applicable inspection requirements outlined in Section 02 of this procedure. The expenditure of inspection resources and the emphasis on inspection activities should be based on the scope of the replacement activities and associated modifications, the safety significance of the activities, the licensee's historical performance in that area, and industry experience. Additional inspection emphasis may be considered for those aspects of the replacement project that include new or different management controls or involve new techniques. For example, there may be new administrative procedures or quality assurance programs for the project or specialized welding techniques such as the "narrow gap" method.

The inspection requirements in Section 02 of this procedure are general and address the basic activities that should be inspected. Additional guidance for specific inspections should be derived from inspection procedures relative to the area being examined. Those procedures include, but are not limited to, the procedures listed in Appendix A, "Applicable Inspection Procedures," of this procedure.

NOTE: Some of the procedures listed applied to original construction activities, preoperational testing, and startup testing and may be used as guidance for inspection of SG replacement.

Specific Guidance

03.01 Inspection Scoping

The three major phases of a SG replacement project are typically design and planning, SG removal and replacement, and post-installation verification and testing. The licensee's design and planning phase may begin several years before replacement. Contracts for SG procurement have been issued three or more years before replacement. The licensee may install temporary or permanent modifications to support the SGRP well in advance of the SGRP outage. Significant design work and outage planning may be complete a year before the outage.

The inspection should focus on SGRP activities that verify proper restoration of pressure boundaries of the reactor coolant system (RCS), secondary systems, and containment systems, exclusion of foreign materials, and plant modifications that could affect plant risk during subsequent plant operation

The inspector should contact the Office of Nuclear Reactor Regulation (NRR) project manager to obtain any specific technical input related to the inspection from NRR licensing review of any licensee SGRP submittal

An early review of the licensee's SGRP scope and schedule will provide advance opportunity to identify special inspection needs and plan the design and planning phase inspections. For the significant support modifications installed prior to the outage, inspection before the SGRP outage is recommended. About a year before the SGRP outage, the licensee's outage scope and schedule should be detailed enough to develop the inspection plan for review of onsite work. Inspection planning has typically been performed by a region-based inspector with input from the resident inspector.

The scope and depth of the inspection of these parts and the inspection hours credited to the baseline inspections should be in accordance with the guidance in this procedure. Inspection planning for the baseline inspections should consider the extended schedule for the SGRP and the prudent management of baseline inspection hours.

03.02 Design and Planning Inspections

- a. Inspection samples of permanent and temporary plant modifications for the SGRP should be selected and reviewed as a part of the baseline inspection using applicable portions of 71111.21M, "Comprehensive Engineering Team Inspection," and IP 71111.18, "Plant Modifications."

10 CFR 50.59 evaluations and screening for such evaluations for selected modifications related to SGRP should be reviewed using applicable portions of IP 71111.21M.

The inspection should verify that (1) key SG design aspects such as those listed below, SG modifications, and the designs of other related significant modifications are reviewed and approved in accordance with procedures and (2) replacement materials and components meet the appropriate design technical requirements. Those technical requirements include the applicable codes and standards, NRC requirements, and other commitments made by the licensee in the FSAR.

The inspection should review design attributes like those outlined in IP 71111.21M.

The following list illustrates typical topics that may need to be reviewed:

1. The general design for removal and reinstallation of the SGs and related components.
2. If needed, the design and analysis for the creation of a temporary containment opening for SG replacement and its restoration - design reviews should be coordinated with the **Structural, Civil, Geotech Engineering Branch**, NRR, via the Project Manager, to confirm the restored design margin - see Appendix B of this procedure for additional information.
3. The impact of changes in mass and center of gravity of the new steam generator on the seismic analysis for the containment structure, pipe stress analysis, and other safety systems and components.
4. The effect of the **SG** and related design changes on transient and accident analyses including tube ruptures. For example, if the secondary side volume for water/steam is reduced, or the power generation rate is increased, the secondary side flow rate may increase, creating the need to evaluate new flow rate vibration drivers and limitations, to prevent excess tube vibrations.

NOTE: When the need for specialized knowledge or experience is identified, such as for an evaluation of flow vibration, the resident and/or regional inspection staff should coordinate the review through the NRR Project Manager for SG design technical expertise for this aspect of the inspection.

5. The cumulative and synergistic effects, if any, of **SG** related design changes, and other modifications completed during the outage on transient and accident analyses. For example, the cumulative and synergistic effects of design changes, if any, on the susceptibility of the **SG** tubes to flow-induced vibration, and whether the changes could result in operation outside the envelope of successful previous operating experience. See NUREG-0800 "Standard Review Plan," Section 5.4.2.1, "Steam Generator Materials and Design," for further information.
6. Adherence to Construction/Fabrication and reconciliation of Construction Code requirements.
 - (a) Using guidance from Appendix D, "**Head Fabrication & Preservice Record Review Checklist**," of IP71007, "**Reactor Vessel Head Replacement Inspection**," confirm that fabrication records demonstrate Code Compliance related to: SG vessel & tube material & welds—**Certified Material Test Reports**, heat treatment, forging toughness, fabrication NDE, and resolution of repairs and vendor nonconformances.
 - (b) Verify that the Design Specification is reconciled or **updated**, and a Design Report is prepared for the reconciliation of the **replacement SG (RSG)**. Both Design Specification and Report should be certified by professional engineers competent in **American Society of Mechanical Engineers (ASME)** Code requirements.
7. Verify that preservice NDE requirements were completed for the RSG welds and tubes in accordance with the **ASME Boiler & Pressure Vessel Code, Section XI**.

8. Impact of the RSG modification on the residual heat removal (RHR) system and auxiliary feedwater (AFW) system cooldown performance—design basis cooldown/heat load calculation and minimum required inventory in storage tanks supporting cooldown with RHR or AFW. Typically, an increased metal mass of the RSG increases the heat load for these systems, which in turn increases the minimum required water inventory for the storage tanks which support plant cooldown. **Verify that the design calculations are based on the power uprated conditions if the NRC has approved the unit for the power uprate.**
9. Impact for the containment sump level and pH levels for a Main Feed Line Break (MFLB) and Main Steam Line Break (MSLB) with the RSGs. Sump volume may increase, PH may decrease, and containment spray systems may actuate in the event of a MFLB with the RSG because of the increased volume and/or energy releases.
10. Margin between the RCS mechanical design flow and the RCS best estimate flow for the RSG modification. RSG may have less flow resistance resulting in an increased pressure drop across the reactor core.
11. Impact of increased RCS volume effecting other RSG or accident analyses. RSGs typically contain more tubes which can increase the RCS volume and impact many analyses.
12. Impact of RSG internal main steam line flow restrictor cross sectional area and tube surface area on the MSLB analyses. Typically, the RSG has increased tube surface area and may have a different flow restrictor design, which needs to be given appropriate consideration in MSLB analyses.
13. Impact of RSG on margin to overflow (MTO) analysis. Under a steam generator tube rupture condition, the SG rapidly fills with water from the leaking tube and the automatic initiation of AFW flow. An accident acceptance criterion for this condition is that the RSG will not be overfilled and a MTO is used to describe the volume remaining in the SG prior to reaching a solid condition. Filling the SG solid could induce the SG power operated relief valve or safety valves to stick open resulting in offsite release of radioactivity (i.e., containment bypass). Operator actions to identify and isolate the affected SG are relied on to prevent this overfill condition and offsite release.
14. **Confirm that the location and elevation of the RSG considers the fit-up between the existing pipes and the RSGs nozzles before welding, such that the location is within the design tolerance (clearance). Confirm that the design tolerance is within the stress analyses (including various operating conditions) of the existing pipes and RSGs. Confirm that the fit-up between the existing pipes and the RSG nozzles in the field is within the design tolerance. This is to minimize the stresses on the existing pipes and RSG nozzles.**
15. **If differences exist between the RSGs and old SGs in the areas of (1) dimensions (outside diameter, height, and support skirt) and, (2) the nozzle locations on SGs, the inspector needs to verify whether the licensee has performed new stress analyses or updated the existing stress analyses of the attached pipes. In any event, the inspector needs to verify that the existing/updated/new stress analyses of the pipes attached to RSGs are applicable after RSG installation (i.e., the resulting stresses at**

the RSG nozzles and attached pipes are applicable and are within the allowable stresses of the ASME Code, Section III, and the analytical model represents the RSGs.) The inspector needs to verify that the loads on RSG nozzles do not deviate excessively (more than 5 percent) from the nozzle loads in the existing stress analyses. The inspector should contact the cognizant NRR technical branch.

- b. Several heavy lifting evolutions are expected. While the radiological consequences of a heavy load drop are expected to be small with the core defueled, a dropped component could result in an unwarranted radioactive release and severe damage to equipment needed for reactor or radiological safety.

The licensee's plans and analysis for lifting and rigging of heavy loads are reviewed to verify that the safe load path analysis for component removal and reinstallation is technically sound. The inspection should focus on the impact of heavy load lifting operations on spent fuel and its cooling, support systems for the reactor, and common support systems for the other operating reactor unit(s) at the site.

Modifications such as reinforcement of existing structures and/or floors, construction of new structures or platforms, and changes to the crane should be reviewed to verify that the modified equipment will safely handle expected loads. The potential impact of these modifications on safety-related equipment should be assessed for applicability under 10 CFR 50.59.

The component drop analysis should be reviewed to verify, in general, that the potential offsite releases at the exclusion area boundary are within 10 CFR Part 100, "Reactor site criteria," limits and equipment to maintain safe shutdown will be unaffected.

Additional information on lifting heavy loads is available in Generic Letter 81-07, "Control of Heavy Loads," NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and Regulatory Guide 1.244, "Control of Heavy Loads at Nuclear Facilities." Appendix C, "Supplemental Guidance for Inspection of Lifting and Rigging Activities," of this procedure provides additional inspection guidance associated with lifting and rigging activities.

- c. Specific guidance for radiation protection inspections is provided in IP 71124.01, "Radiological Hazard Assessment and Exposure Controls," IP 71124.03, "In-Plant Airborne Radioactivity Control and Mitigation," IP 71124.04, "Occupational Dose Assessment," IP 71124.06, "Radioactive Gaseous and Liquid Effluent Treatment," and IP 71124.08, "Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation."

The inspection should audit the licensee's outage radiation protection program and confirm that radiological concerns are factored into SGRP planning. When applicable, review of radiological concerns associated with the establishment and restoration of a temporary containment opening should be included.

For exposure and contamination control issues, attention should be directed to potential high dose and/or high contamination activities such as reactor coolant system piping cutting and welding or work in or around the SG lower channel head. Where applicable, plans for shielding installations should consider the effects of additional weight loads on equipment.

Training for the large SGRP work force should be geared toward the specifics of SG removal/replacement for both the radiation workers and radiation protection personnel. Review of planning for radioactive materials management should be considered due to the need for temporary storage of reusable equipment and the handling of the increased volumes of waste generated in a short time. Efficient removal of materials from the work area improves housekeeping and reduces fire hazards. Emergency contingencies such as actions for a dropped component and the plans for closure of the containment (including temporary openings) in a high airborne condition should be reviewed.

In the past, licensee incorporation of lessons learned from earlier SGRPs has resulted in improvements in SGRP radiation protection planning. Refer to NUREG/CR-1595, "Radiological Assessment of Steam Generator Removal and Replacement Update and Revision," for additional information on radiological considerations.

- d. Security boundaries may be affected during the SGRP. The impact of the work on the boundaries and the licensee's contingency plans should be reviewed to verify conformance with the security plan.

03.03 Steam Generator Removal and Replacement Inspections

- a. Significant cutting, machining, and welding activities are expected with SG replacement outages. Several procedures are listed in Appendix A to this procedure that contain useful guidance for welding inspections.

Training and qualifications of licensee and contract quality control/assurance inspectors, and NDE examiners are reviewed to verify that personnel meet site and ASME Code qualification requirements and are prepared for the site-specific tasks. In the past, mockups have been used for qualifying welding procedures and training and qualifying welders, machinists, and NDE inspectors.

For the selected welds review weld procedures and welder qualification records, confirm that the ASME Code required essential and supplemental essential welding variables for the welding processes used were met, and verify that preservice NDE requirements were completed with acceptable results in accordance with the ASME Code, Section III. Also, review the non-conformance reports for each major steam generator replacement weld to confirm that welding deficiencies were dispositioned in accordance with Code requirements.

- b. In many previous SG replacement activities, the licensee used nickel-based Alloy 182 as the weld filler material to connect major inlet and outlet pipes (hot and cold leg) to the RSG nozzles. The licensee also applied a weld inlay on the inside diameter surface of the pipe using Alloy 52/52M weld filler metal at the attachment nozzles to insulate the Alloy 182 weld metal from RCS coolant to minimize primary water stress corrosion cracking in Alloy 182 weld metal.
- c. The inspector should verify that the appropriate weld material and welding procedure are used to connect the existing pipes to RSG nozzles. Additionally, the inspector should determine if the licensee performed a repair of the finished weld after an unacceptable fabrication defect was detected by fabrication or preservice examinations as required by ASME Code Section III.

- d. If a weld repair was performed, the inspector should (1) verify whether the licensee repaired the weld in accordance with the ASME Code Sections III and XI, (2) document the repair activity and the existence of the fabrication defect in the inspection report, (3) document the results of nondestructive examinations of the welds between the existing pipes and RSG nozzles in the inspection report, and (4) communicate the issues to the cognizant NRR technical branches via the NRR project manager.
- e. The inspection should verify that modifications to walls and other structures and removal and restoration of component supports are documented.
- f. A temporary containment opening may need to be established to allow for the movement of SG components. The inspections should verify that the procedures and methods used to enlarge and restore the containment access ensure that the design assumptions and requirements are satisfied, and that the containment is restored to at least its required strength and integrity criteria. Post restoration testing should demonstrate that the containment has been properly restored. See Appendix B of this procedure for additional information.
- g. Implementation of controls for personnel access to radiologically controlled work areas, surveillance of work activities, and procedure adherence should be verified. Reviews should be made of the implementation of ALARA, radiological exposure, contamination, and airborne contamination controls planned for cutting, welding, and other activities including contaminated interference removal. Also, implementation of any special controls for contaminated tools and waste should be reviewed. Where applicable, controls for shielding installations should consider the effects of additional weight loads on equipment.

During SG change out activities, the licensee will be conducting cutting, decontaminating, and welding activities on highly contaminated primary piping systems. Loss of normal ventilation controls, including portable ventilation system integrity, may result in elevated airborne radioactivity. Since containment openings are not recognized as evaluated effluent release pathways in the Offsite Dose Calculation Manual (ODCM), the licensee should establish robust barriers to preclude inadvertent airborne effluent releases. The controls should preclude, identify, and promptly correct effluent control issues (e.g., inward airflow should be maintained, means to timely detect outward airflow and loss of negative flow, curtains or barriers should be in place and remain closed except when in use). The licensee should validate that **any** multiple openings are not creating local chimney type effects that could serve as release paths. In addition, monitoring at openings should be in place to detect and quantify potential radioactive releases at an appropriate sensitivity for purposes of both effluent release evaluation as well as emergency declaration (e.g., Unusual Event).

If a temporary containment opening is established, the method of disposal of any concrete debris may be considered. While much of this material is not expected to be contaminated, consideration should be given to any activation materials including hard to detect isotopes such as H-3 and Fe-55.

Review the licensee's evaluations of adverse impact of temporary services (electrical power, structural supports, and systems) on plant equipment and systems important to safety.

- h. If the **old** SGs are stored on site, the storage facility should be reviewed to verify that access is properly controlled and dose rates at the perimeter are below applicable limits. For additional information, refer to Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites."
- i. **No specific guidance**

03.04 Post-installation Verification and Testing Inspections

The inspection should review the licensee's post-installation verification and testing program to verify that modifications are completed in accordance with the design; that drawings, procedures, and training have been updated as appropriate; that post-installation walkdowns and inspections are performed to ensure that equipment is restored and temporary services are removed; that RCS piping-to-restraint clearances are verified; that equipment cleanliness has been verified; that preservice inspections of welds to establish baseline data are performed; and that deficiencies are properly dispositioned. Verify that changes in performance of the SGs and in its associated parameters, such as flow rates, pressures, and temperatures are appropriately included in design documents and plant procedures **and satisfy the current licensing basis**. In addition to reviewing the licensee's program to verify restoration, the inspector may perform independent walkdowns to confirm equipment restoration.

If a temporary containment opening was established for SG replacement, refer to Appendix B of this procedure for additional guidance on testing following restoration.

Direct inspection of portions of the primary and secondary **system** leakage tests and **associated** visual examinations and review of the test results should be performed. The inspection should verify that testing satisfies **the ASME Code, Section XI, IWA-5000**, and applicable regulatory requirements, that testing was conducted according to the procedure, and that results were satisfactory or properly resolved.

Section 02.04 of IP 93803, "**Safety Systems Outage Modifications Inspection**," contains additional inspection guidance for testing inspections.

50001-04 INSPECTION RESOURCES

The required number of direct inspection hours per SGRP plant will vary significantly based on **several** factors such as licensee experience, contractor experience, and the variable complexities of the given SG replacement. The scope and depth of the inspections should be sufficient to provide the desired level of assurance that the licensee adequately performs activities important to safety and that the relevant codes, standards, requirements, and commitments are met.

The total resources required for SGRP inspection using IP 50001 is estimated to be 350 direct inspection hours.

NOTE: It is expected that portions of the inspections in this procedure will be credited towards satisfying baseline inspections, such as IPs 71124.01, 71124.03, 71124.04, 71124.06, 71111.18, and 71111.21M. A goal of approximately 110 hours at a single unit site and 140 hours at a multi-unit site should be credited towards baseline inspections (as allowed by inspection schedule circumstances). Inspection

of the establishment and/or restoration of a temporary containment opening will require additional inspection effort.

50001-05 REFERENCES

ASME Boiler & Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components (See 10 CFR 50.55a, "Codes and standards")

ASME Boiler & Pressure Vessel Code, Section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators"

ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants"

Generic Letter 81-07, "Control of Heavy Loads"

Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites"

NUREG/CR-1595, "Radiological Assessment of Steam Generator Removal and Replacement Update and Revision"

NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" (ML070250180)

NUREG-0800 "Standard Review Plan," Section 5.4.2.1, "Steam Generator Materials and Design"

Regulatory Guide (RG) 1.136, Revision 4, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments" (ML20301A167)

RG 1.244, Revision 0, "Control of Heavy Loads at Nuclear Facilities" (ML21006A346)

RG 1.57, Revision 2, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components" (ML12325A043)

END

Appendices:

- A. Applicable Inspection Procedures
- B. Temporary Containment Opening Review Guidance
- C. Supplemental Guidance for Inspection of Lifting and Rigging Activities

Attachment 1: Revision History for IP 50001

Appendix A: Applicable Inspection Procedures

Inspection Procedure No.	Inspection Procedure Title
71111.08	Inservice Inspection Activities
71111.18	Plant Modifications
71111.21M	Comprehensive Engineering Team Inspection
71124.01	Radiological Hazard Assessment and Exposure Controls
71124.03	In-Plant Airborne Radioactivity Control and Mitigation
71124.04	Occupational Dose Assessment
71124.06	Radioactive Gaseous and Liquid Effluent Treatment
71124.08	Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation
55050	Nuclear Welding General Inspection Procedure
55100	Structural Welding General Inspection Procedure
57050	Nondestructive Examination Procedure Visual Testing Examination
57060	Liquid Penetrant Testing Examination
57070	Magnetic Particle Testing Examination
57080	Procedure Ultrasonic Testing Examination
57090	Nondestructive Examination Procedure Radiographic Examination Procedure Review/Work Observation/Record Review
88143	Pipe Supports and Restraints
93803	Safety Systems Outage Modifications Inspection

	Containment Inspections
62003	Inspection of Steel and Concrete Containment Structures at Nuclear Power Plants
70307	Containment Integrated Leak Rate Test Procedure Review
88132	Structural Concrete Activities
70313	Containment Integrated Leak Rate Test Surveillance
70323	Containment Leak Rate Test Results Evaluation

END

Appendix B: Temporary Containment Opening Review Guidance

A. OBJECTIVE

This appendix provides additional guidance for inspection of activities associated with establishing a temporary containment opening in the containment shell for steam generator (SG) replacement and its subsequent restoration.

B. BACKGROUND

To replace the SGs in containments that have small equipment hatches, some licensees have chosen to replace the SG in one piece through a temporary transfer opening cut through the containment instead of disassembling the SG and transporting its sections and components through the equipment hatch. Based on time and radiological considerations, this approach has generally been preferable. After completion of the SG replacement, the opening is closed, the containment structure is restored to its original configuration and integrity, and an integrated leak rate test (ILRT) is performed. Cutting the temporary opening and closing it may be carried out without NRC's approval under the provisions of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.59. "Changes, tests and experiments," (See the table in page 52 of NUREG-1379, Rev. 3.) provided the process does not meet the criteria contained in 10 CFR 50.59(c)(2) or require a change in technical specifications. However, since this process involves construction activities for the most important safety-related structure in a nuclear power plant, quality assurance and quality control practices should be strictly followed, and the restoration of the containment should be inspected by the NRC.

C. GUIDANCE

General

This appendix is intended to augment the existing inspection guidance in this procedure to identify regulatory concerns and licensee activities associated with cutting and closing a temporary containment opening. Some of the discussion below may overlap previous guidance and is only provided for clarity.

Nuclear Regulatory Commission inspection emphasis should be on verification that the containment is restored to its original integrity after closure of the temporary opening. Because actions related to cutting and restoring a temporary opening in a prestressed concrete containment encompass issues concerning reinforced concrete and steel containments, issues related to prestressed concrete containments are used as the basis for this appendix. Therefore, some items discussed below may not apply to all containment designs.

Construction as used herein is an all-inclusive term that covers materials, design, fabrication, installation, examination, and testing. The licensee should prepare specifications covering all aspects of the construction of the temporary opening as discussed in the following sections.

Regulatory Concerns

In a prestressed concrete containment, the components that are cut and replaced include concrete, reinforcing bars, tendon sheathing, and the steel liner. The tendons, together with the grease in the sheathings that pass through the opening area, are removed before the opening is cut. The tendons may be reused in the restoration. The concerns are as follows:

1. The steel liner plate (or metallic shell for steel containments) may be torch-cut and reused. During reinstallation, the gap around the liner plate for welding may not be as uniform as it would be for new plates. The uniformity of the gap is dependent upon how skillfully the plate was initially torch-cut. The strength of the replacement welds may not be the same as the uncut plate, but it should be very similar to the strength of the weld joints in the original structure. If the strength of the replacement welds will be less than the uncut plate, the reduction in weld joint strength should be evaluated as an engineering design change.
2. The creep and shrinkage of the concrete used to close the opening, which is usually high for newly poured concrete, could result in higher compressive strain in the concrete adjacent to the repaired opening because of prestressing and may cause liner buckling.
3. The creep and relaxation of reused tendons could cause the loss of prestress because of tensioning and detensioning.
4. The leak tightness of the tendon sheathing may be compromised because of withdrawing and re-inserting the tendons, resulting in grease leakage. This leakage may reduce the strength of the concrete and leave the tendon unprotected.
5. When containment boundary concrete replacement includes having the reinforcing bar splices lined up without staggering, confirm that an engineering-based evaluation has verified that the strength of the containment has not been compromised.
6. After closing the opening, the containment is to be subjected to a containment pressure test and leakage test in accordance with Subsection IWL and Subsection IWE (specifically Articles IWL-5000 and IWE-5000) of the ASME Code Section XI, as incorporated by reference in 10 CFR 50.55a and subject to the applicable conditions in 10 CFR 50.55a(b)(2)(viii) and (ix). It should be noted that, when applying IWE-5000 to Class MC pressure-retaining boundary components, the condition in 10 CFR 50.55a(b)(2)(ix)(J) requires a Type A integrated leak rate test (ILRT) in accordance with 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," following a major containment modification or repair/replacement activity (e.g., construction openings for replacement of RV head, steam generators or pressurizers). Proposed alternative(s) to these requirements may be requested by the licensee and authorized by the NRC pursuant to 10 CFR 50.55a(z).

Design Reviews

Design reviews should be coordinated with the Structural, Civil, and Geotech Engineering Branch (ESEB), the Office of Nuclear Reactor Regulation (NRR), via the Project Manager, to confirm the restored design margin of the containment structure.

Codes and Standards

Acceptable options for industry codes and standards to be used for the design, construction, and restoration of the containment access are as follows:

(1) The codes and standards described in the FSAR, consistent with current regulations, as the basis of the facility operating license are to be utilized.

Or preferably,

(2) To the extent practical, the latest **ASME** Boiler and Pressure Vessel Code (ASME Code) edition and addenda. Practical adjustments can be made in the use of the updated ASME **C** Code provisions to accommodate the limitations of design and construction. The quality and overall margins required in the original design are to be maintained and should be included in decisions on the extent to which adjustments are made for using the updated ASME **C** Code provisions. If the licensee chooses this option, review of its acceptability should be coordinated with the **(ESEB)** via the Project Manager.

Analysis

The licensee should perform structural analyses for the containment, which considers the effect of the opening for the applicable loads and load combinations resulting from the containment opening.

Structural analyses of the containment structure **should** be performed considering both the presence of the temporary opening and conditions after its repair. This analysis should incorporate the properties of the materials of the replacement components, which may affect the behavior of the containment structure locally and globally. The analyses should demonstrate that the **ASME** Code allowable limits have not been exceeded in both cases. The loads and loading combinations **for the restored configuration (post-opening closure) of containment** should be the same as those specified in the FSAR or the applicable Standard Review Plan sections. Also, consideration should be given to the conditions arising from the existence of the temporary opening, ranging from initial cutting to closing, including potential construction loads. **For the interim configuration with temporary opening of containment, no design pressure (P_a) load associated with the design basis accident will apply in the analysis.**

Design Changes

The materials and replacement components used to restore the containment should be **like** those originally used and documented in the FSAR except that the concrete used to close the opening can be adjusted to reduce the creep and shrinkage. If different materials or components are used, they should satisfy the applicable design criteria and requirements and provide an equivalent overall safety margin as provided in the originally licensed design.

Temporary Opening Inspections

Based on the regulatory concerns previously described, the areas of inspection interest can be summarized as follows:

1. Welding of the steel liner or shell since the torch cut may result in irregular edges. The welding process, use of proper materials, the skill and training of the welder, and post-installation examination and testing should be considered. The replacement weld strength should be confirmed to be compatible with the containment design and it should be very similar to the strength of the weld joints in the original structure.
2. Proper concrete mix design and qualification and proper placement techniques should ensure that concrete in the repair area exhibits low creep and shrinkage. The certification of the concrete mix design should ensure that the design mix results in properties **like** that used in original construction, and that the mix is controlled to reduce creep and shrinkage. Sufficient testing should have been done prior to placement to ensure that design strength is attainable. **The documentation of the certification of materials used in the concrete containment, such as aggregate,**

cement, water, and admixtures **should be reviewed to ensure they meet specified quality requirements.**

Witnessing the initial concrete placement is important to verify that the concrete is being handled and tested properly. In cases where the interior form for the concrete containment is the liner plate, additional stiffeners may be needed to prevent buckling.

3. The tensioning force of the tendons, which may be reused and tensioned to a level of preload which would not cause more creep and relaxation. The prestressing sequence should also be carefully implemented.
4. The control of grease filling so that the tendons are adequately covered and there is no leakage from the sheathing that can be observed on the surface of the containment.
5. The rebar splicing around the opening and the use of the sister splices for quality control testing. Checks on the proper certification of the rebar and splice materials and worker qualifications may also be included.
6. Witnessing of the **containment pressure test or the** integrated leak rate test (ILRT) of the containment.

Fabrication, Installation, Examination, and Testing

Fabrication, installation, examination, and testing with appropriate instrumentation are to be performed in accordance with the applicable ASME Code as stated above under “Codes and Standards,” or the stipulations in the FSAR. The behavior of the repaired containment structure locally and globally as obtained from the **containment pressure test or the** ILRT is to be assessed.

Generally, the facility technical specifications require periodic examinations of the containment integrity, including inspections of the tendons. It is not expected that those examinations will be altered by the restoration of the temporary containment access. Of concern however is the potential for degradation of the containment structural integrity or leak tightness that could result from the restoration. For example, tendon degradation could occur due to moisture intrusion via a pathway created in a damaged or inadequately restored tendon duct. An assessment should be made of the licensee's measures and/or planned actions (e.g., analysis, testing, examination) to verify that containment integrity is maintained following the restoration. Assistance in performing this assessment should be coordinated with the **ESEB** via the Project Manager.

Inspection and Quality Assurance

Quality assurance and independent (third-party) inspection requirements for cutting and closing the temporary opening are applicable. Verification of these activities should be performed.

Training and Qualification

The training and qualifications of craft and quality assurance personnel may be reviewed as there have been past problems with the use of inexperienced personnel.

END

Appendix C: Supplemental Guidance for Inspection of Lifting and Rigging Activities

A. OBJECTIVE

This appendix provides additional guidance for inspection of activities associated with lifting and rigging activities for steam generator (SG) replacement.

B. BACKGROUND

During the SG replacement, several heavy lifting evolutions are expected. While the radiological consequences of a heavy lift load drop are expected to be small when the reactor core is defueled, a dropped component could result in an unwarranted radioactive release and severe damage to equipment necessary for reactor or radiological safety.

C. GUIDANCE

This appendix is intended to augment the existing inspection guidance in this procedure to identify regulatory concerns and licensee activities associated with the lifting and replacement of SGs. Some of the discussion below may overlap previous guidance and is provided only for clarity. The list provided below contains guidance on the various areas of inspection interest.

1. Has the licensee established safe load paths from the containment building and along the haul route to identify potential interactions with equipment important to safety? Has the licensee performed an evaluation to establish that the safe load path that has been chosen is the most effective method of moving the SGs and has the least potential for interaction between the SGs and safety-related **systems, structures, and components (SSCs)**?
2. Typically, the rigging and lifting for the Old Steam Generators (OSGs) and Replacement Steam Generators (RSGs) inside the containment building will be performed as a planned, engineered lift. Has the licensee adopted an acceptable standard (e.g., ASME B30.2, "Overhead and Gantry Cranes") for the execution of the planned "engineered lift" of the OSGs and RSGs inside and outside of containment?
3. Is the design, testing, inspection, and maintenance of the Outside Lift System (OLS) consistent with the applicable guidelines in **Regulatory Guide 1.244, "Control of Heavy Loads at Nuclear Facilities"** and **ASME NML-1, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities"**? Is the OLS considered "single failure proof" and is the OLS specifically designed to withstand the external events (tornado, wind, seismic, natural gas line explosion, etc.) that are a part of the plant's licensing bases with or without the largest postulated load to be lifted?
4. Has the licensee performed a load test of the OLS consistent with applicable industry standards (i.e., ASME B30.2 and ASME B30.9, "Slings") with the largest postulated load to be lifted? What is the largest postulated load to be lifted and how was it determined? Has the licensee performed full-performance tests with 100-percent of the largest postulated load to be lifted for all speeds and motions for which the system is designed to be performed?
5. Does the licensee design, inspect and test the OLS attachments and rigging in accordance with applicable ASME B30 series standards? Does the OLS attachment

- and rigging meet the applicable requirements of ASME HRT-1, “Rules for Hoisting, Rigging, and Transporting Equipment for Nuclear Facilities”? Will the licensee inspect the rigging prior to use in accordance with approved procedures and will rigging operations be controlled and conducted by highly trained and qualified personnel in accordance with approved procedures?
6. Has the licensee evaluated for a “whiplash” effect a loss of lifted load would have on the OLS? (The whip-lash effect can result from a postulated drop of a load from the OLS that can cause instability of the boom masts in the reverse direction, i.e., the masts will not flip over backwards and impact SSCs.)
 7. Has the licensee trained the crane operators of the OLS in accordance with applicable ASME B30 series standards?
 8. Has the licensee evaluated the potential of the mobile crane with its largest lifted load that will be used during assembly/disassembly of the OLS to overturn or collapse and potentially impact safety-related SSCs on the operating unit as well as the defueled unit?
 9. What are the maximum wind conditions for operation of the mobile crane(s)? How the maximum wind condition for operation determined and what was its basis (e.g., dead weight of the boom with maximum postulated lifted load)?
 10. Does the licensee intend to load test the mobile crane that will be used during assembly and/or disassembly of the OLS?
 11. How has the licensee minimized or eliminated conditions that could result in credible crane failure modes or load drops (i.e., operator errors, use of improper rigging or inappropriate slings, and crane component failures)?
 12. Review the description of the OLS foundation and how the OLS is anchored to the foundation. Has the OLS foundation been analyzed to withstand the imposed loads associated with the rigging operation (lifted load) including dead, live, wind, and seismic?
 13. What are all the safety-related SSCs on the operating unit (if applicable) as well as the defueled unit that can be adversely affected by the drop of a SG? If testing of the OLS will occur onsite, will that introduce additional safety-related SSCs on the operating unit (if applicable) as well as the defueled unit that can be adversely affected by the drop of a SG in addition to the SSCs affected by drop of SG during the replacement activities?

Example of safety-related SSCs but not limited to:

- Both trains of underground diesel fuel oil supply piping to the Emergency Diesel Generator Day Tanks.
 - Underground power to both trains of the diesel fuel oil transfer pumps.
 - Underground Service (raw) water piping to/from the ultimate heat sink.
 - The underground firewater system piping.
14. How will the licensee verify during and following erection of the OLS that the proper assembly of electrical and structural components took place?

15. Does the licensee have a heavy **load** lift plan detailed in engineering packages, which defines the requirements for the safe rigging of the heavy loads associated with the SG Replacement project?
16. How will the licensee demonstrate the ability of the OLS to protect against an overload situation and to include the ability of the OLS to withstand a load hang-up?
17. What are the heavy load drop protection plans and compensatory measures?
18. What is the maximum wind speed allowed during operation of the OLS (when the lifted load is off the ground and outside the containment) in any direction measured at the boom tip? Does the licensee have adequate controls in place to limit operation of the system with presence of adverse wind condition as necessary?
19. Does the licensee have procedures in place to delineate specific actions required in case of a heavy load drop? When will the procedures be completed, who will require training on these procedures, and how far in advance will training be completed relative to heavy lift operations?
20. Do the interfacing lift points on the old/new SGs, such as the lifting lugs meet the applicable guidelines of NUREG-0612, Section 5.1.6(3)(a) or (b)? What criteria are the interfacing lift points (i.e., the SG trunnions) designed to meet?
21. The OLS is typically a commercial crane not specifically designed for use at a nuclear facility. Has the license incorporated operating experience (Arkansas Nuclear One 2013 event, Miller Park 1999 event) to take lessons learned and incorporate the applicable insights into their heavy lift plan?
22. Will cranes (OLS and mobile cranes used to erect the OLS) and work areas be equipped with strategically located instruments to monitor wind velocity (speed and direction) at or near the elevation of lifted loads?
23. Has the licensee evaluated the temporary increase in risk during the time the SGs pass over safety-related SSCs **in accordance with the guidance of RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"**? This increase in risk is **like** the increase in risk associated with other planned, temporary plant activities (i.e., maintenance on safety-related equipment covered by the **technical specifications**).
24. What components are included in the weight of the lifted loads? Review the loads to be lifted and whether the lifted loads are calculated or estimated. What means will be used to verify the weight of the lifted loads in the field?
25. Are the lifting devices used for the lifting of SGs designed, inspected, and tested with the applicable ASME series standards? What criteria are the lifting devices designed to meet?

26. In accordance with recommendations provided in NUREG-0612, Section 5.1, review how the licensee addressed the potential for accidental dropping of the steam generator inside the reactor containment building. Review the potential consequences that could result from dropping the steam generator and any compensatory measures that could be implemented to minimize and manage the damage from the drop.

END

D. REFERENCES

American Society of Mechanical Engineers (ASME) B30.2, "Overhead and Gantry Cranes"

American Society of Mechanical Engineers (ASME) B30.9, "Slings"

American Society of Mechanical Engineers (ASME) HRT-1, "Rules for Hoisting, Rigging, and Transporting Equipment for Nuclear Facilities"

American Society of Mechanical Engineers (ASME) NML-1, "Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities"

Regulatory Guide (RG) 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications"

RG 1.244, "Control of Heavy Loads at Nuclear Facilities"

END

Attachment 1: Revision History for IP 50001

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
N/A	ML080250198 08/19/08 CN 08-024	Updated referenced procedures to reflect ROP Realignment changes issued 1/31/08 and ROP feedback forms regarding resource planning, incorporating 71122.01 as a reference, removing SIT while keeping ILRT throughout the procedure, and various technical and editorial changes.	N/A	ML082190966
N/A	ML082620073 03/23/09 CN 09-010	Deleted Option A from Appendix B, Part C, Paragraph f and added a subsequent note for clarification. Removed inspection requirement 02.02.a.3, clarified the Inspection Resource note, and made other minor changes.	N/A	ML083640218
N/A	ML11206B197 11/08/11 CN 11-031	Revised to incorporate feedback associated with Feedback Form No. 50001-1635. For historical reference, ADAMS Accession number for previous revision is ML082620073.	N/A	ML112840062
N/A	ML17083B311 09/12/17 CN 17-018	Addressed Feedback Forms 50001-2027, 2032, 2223, and 2244.	N/A	ML17157B192 50001-2027 ML17206A151 50001-2032 ML17178A233 50001-2223 ML17178A248 50001-2244 ML17206A154

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
N/A	ML22201A513 02/04/25 CN 25-001	Researched commitments for 4 years and found none. Minor editorial changes. This completes the 5-year review.	N/A	ML22213A211