

# **NRC INSPECTION MANUAL**

DANU/UARP

---

## **INSPECTION MANUAL CHAPTER 2573**

---

**INSPECTION OF THE ADVANCED POWER REACTOR  
“QUALITY OF REACTOR PLANT CONSTRUCTION”  
STRATEGIC PERFORMANCE AREA**

**Effective Date: 02/05/2026**

## TABLE OF CONTENTS

2573-01	PURPOSES .....	1
2573-02	OBJECTIVES .....	1
2573-03	APPLICABILITY .....	2
2573-04	DEFINITIONS.....	2
2573-05	RESPONSIBILITIES AND AUTHORITIES.....	4
2573-06	REQUIREMENTS .....	5
2573-07	GUIDANCE .....	5
07.01	General.....	5
07.02	Design-Specific Matrix.....	6
07.03	Project-Specific Matrix.....	7
07.04	Nth-of-a-Kind (NOAK) Inspection Scoping .....	7
a.	Reactor Plant Sites (i.e., the site where the reactor will eventually operate).....	7
b.	Manufacturing and Project Vendor Facilities.....	7
07.05	Inspection .....	8
07.06	Inspection Planning .....	9
07.07	Inspection Sample Selection .....	9
07.08	Inspection Results .....	9
07.09	Assessment of Inspection Results .....	9
07.10	Enforcement .....	9
2573-08	REFERENCES .....	10
Attachment 1:	IMC 2573 Inspection Procedures .....	Att1-1
Attachment 2:	Design-Specific Inspection Scoping Matrix.....	Att2-1
Background:	.....	Att2-1
Developing The Design-Specific and Project-Specific Matrices:	.....	Att2-2
Management Review:	.....	Att2-5
Attachment 3:	Response to Significant Issues or Events .....	Att3-1
Attachment 4:	Revision History for IMC 2573 .....	Att4-1

2573-01        PURPOSES

- 01.01 To provide instructions and guidance for implementation of the Advanced Reactor Construction Oversight Program (ARCOP) for inspections in the “Quality of Reactor Plant Construction” strategic performance area, including guidance for scaling project-specific inspection scopes.
- 01.02 To provide a basis for identification of the minimum set of inspection requirements under the “Quality of Reactor Plant Construction” strategic performance area that provides reasonable assurance that a reactor facility has been constructed in accordance with its licensing basis.
- 01.03 To provide guidance for establishing a minimum baseline inspection program for project vendors and manufacturers for nth-of-a-kind (NOAK) microreactors or significant portions of reactor modules.

2573-02        OBJECTIVES

- 02.01 To develop, in accordance with Attachment 2 of this IMC, design-specific inspection scoping matrices (design-specific matrices).
- 02.02 To identify in the design-specific matrix the inspections, tests, analyses, and acceptance criteria (ITAAC) applicable for plants licensed under 10 CFR Part 52.
- 02.03 To develop, in accordance with Attachment 2 of this IMC, project-specific inspection scoping matrices (project-specific matrices) from the applicable design-specific matrix by adding site-specific structures, systems, and components (SSCs) and ITAAC to the design-specific matrix; and scaling the project-specific baseline inspection program using construction and operating experience (ConE and OpE).
- 02.04 To allow modification of project-specific matrices for subsequent and nth-of-a-kind (NOAK) reactors based on prior inspection results.
- 02.05 To provide guidance for implementing the project-specific matrices in the most effective and efficient manner based on specific advanced reactor manufacturing and construction deployment models.
- 02.06 To verify licensees and project vendors have quality fabrication, construction, and manufacturing programs, processes, and procedures; provide adequate quality assurance (QA) oversight of fabrication, manufacturing, and construction activities; and identify and correct conditions adverse to quality.
- 02.07 To describe the baseline inspection program elements for project vendors and manufacturers, including transition to periodic inspections once the NRC has reasonable assurance the project vendor’s or manufacturer’s ability to produce quality microreactors or significant portions of reactor modules.

## 2573-03 APPLICABILITY

- 03.01 This IMC is applicable to the fabrication, manufacture, and construction of advanced power reactors. Advanced power reactors include all commercial small modular reactors (SMRs), microreactors incorporating both light water reactor (LWR) and non-LWR technologies, and large commercial reactors with additional inherent safety features. Activities under this IMC may begin when an application for the manufacture or construction of an advanced power reactor facility has been accepted and docketed by the NRC. This includes applications for a construction permit (CP), limited work authorization (LWA), combined operating license (COL), or manufacturing license (ML).
- 03.02 This IMC shall be coordinated with the vendor inspection program (VIP) to ensure ARCOP and VIP inspection scopes do not overlap and that both licensed manufacturers (ML holders) and non-licensed project vendors are clearly scoped into the ARCOP.
- 03.03 This IMC is no longer applicable to a reactor or reactor plant once the reactor or reactor plant has been transferred to the operating reactor oversight process (ROP).

## 2573-04 DEFINITIONS

Applicable ARCOP definitions are in IMC 2570, "Advanced Reactor Construction Oversight Process General Guidance and Basis Document." For readers' convenience, some relevant definitions are also listed below.

- a. **Advanced Reactor Construction Project**. The fabrication, manufacturing and construction of one or more advanced commercial reactors intended to be operated by the same licensee at a common location. A reactor construction project includes fabrication activities performed at a non-licensed project vendor facility, reactor manufacturing activities at a manufacturing facility, and reactor construction at its intended operating site, as applicable.
- b. **ARCOP Information Management System (AIMS)**. An information technology platform used to aid in planning, implementing, and tracking advanced reactor fabrication, manufacturing and construction inspections.
- c. **Design-Specific Inspection Scoping Matrix**. A matrix that includes the quality of reactor plant construction strategic performance area portion of the baseline inspection plan for a specific advanced reactor design. The design-specific matrix identifies the safety-related and safety-significant SSCs for a specific design (i.e., matrix rows) and the inspection areas (i.e., matrix columns) that are applicable to the reactor plant design. AIMS is typically used to create, save, and edit design-specific matrices.
- d. **Direct Observation Techniques**. Direct observation includes observing in-process fabrication, construction or manufacturing-related activities such as qualification, assembly, installation, inspection, examination, and testing to determine if the activity was performed in accordance with the licensing basis and appropriate work control documents (e.g., applicable instructions, procedures, and/or drawings).
- e. **Fundamental Safety Functions (FSFs)**. A set of high-level functions that serve to limit the release of radioactive materials to within established limits over the entire range of licensing basis events. FSFs are discussed in various references, such as in Nuclear

Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," (endorsed by Regulatory Guide 1.233). The FSFs are common to all reactor designs and therefore provide a technology-inclusive oversight framework. The FSFs are:

1. Reactivity Control,
  2. Control of Heat Removal (including reactor and spent fuel decay heat and heat generated from waste stores), and
  3. Radionuclide Retention.
- f. **Project-Specific Inspection Scoping Matrix.** A matrix that includes the quality of reactor plant construction strategic performance area portion of the baseline inspection plan for a specific reactor project. Project-specific matrices are typically created in AIMS using the design-specific matrix as a starting point, updated based on construction and operating experience from previous deployments, and then adjusted to include applicable site-specific safety-related and safety-significant SSCs/ITAAC.
- g. **Project Vendor.** A non-licensed entity that fabricates nearly complete reactor plants or significant portions of safety-significant system modules under contract to an NRC licensee, NRC permit holder, or an applicant for an NRC license or permit.
- h. **Reactor Manufacturer.** An ML holder that produces complete reactor plants (e.g., microreactors), or nearly complete reactor plants (e.g., SMR power modules). A reactor manufacturer may produce reactors for multiple reactor construction projects.
- i. **Record Review.** Record review includes review of a sample of completed quality records to determine if the fabrication, construction or manufacturing-related work activity was performed in accordance with the licensing basis and the appropriate applicable instructions, procedures, and drawings. For the records reviewed, inspectors should determine if records are adequate to furnish evidence that activities affecting quality were performed correctly. If possible, the inspectors should also perform a walk-down of the completed work activity associated with the records reviewed to determine whether the as-built SSC conforms with the final design, fabrication, manufacturing and construction documents, and the records reviewed.
- j. **Traditional Vendor.** For the purposes of this manual chapter, traditional vendors are non-licensed entities that supply basic components such as material, equipment, components, or services to a project vendor, manufacturer, licensee, or applicant to be used in an NRC-licensed facility or activity. In certain cases, the vendor may be an NRC licensee (e.g., a nuclear fuel fabricator) or the product may have NRC certificates (e.g., a transportation cask). Traditional vendors may include suppliers of basic components or third-party commercial grade dedicating entities. Traditional vendors are inspected under the VIP. See the definition for "project vendor" for an explanation of the difference between a project vendor inspected under the ARCOP and a traditional vendor inspected under the VIP.
- k. **Vendor Inspection Program (VIP).** The NRC inspection program that verifies applicants and licensees are fulfilling their regulatory obligations with respect to providing effective

oversight of the nuclear supply chain for both operating reactors and new reactor design and construction activities through a strategic sample of vendor inspections.

## 2573-05 RESPONSIBILITIES AND AUTHORITIES

### 05.01 Director, Office of Nuclear Reactor Regulation (NRR)

- a. Provides overall program direction for the inspection of advanced power reactors.
- b. Ensures that NRR includes adequate staff in the ARCOP program organization to carry out the ARCOP.

### 05.02 Regional Administrators / Deputy Regional Administrators (RA/DRA)

Ensures that the Regions include adequate numbers of inspectors in the various disciplines necessary to carry out the ARCOP as described in this IMC.

### 05.03 Director, Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) (NRR)

- a. Acts as the ARCOP program office director (APO Director)
- b. Responsible for the content of this IMC.

### 05.04 Chief, Advanced Reactor Policy Branch (UARP)

- a. Acts as the APO Branch Chief.
- b. Responsible for periodic updates to IMC 2571 in accordance with IMC 0040, "Preparation, Revision, Issuance, and Ongoing Oversight of NRC Inspection Manual Documents."

### 05.05 NRR/DANU Staff - ARCOP Project Organization (APO, NRR)

- a. Administers and monitors the execution of the ARCOP inspection program.
- b. Collects feedback from the construction inspection organization, the host regions, and the VIP, to ensure consistent implementation of the ARCOP inspection program.
- c. Develops and maintains design-specific matrices for each unique advanced power reactor design.
- d. Develops project-specific matrices for each advanced power reactor construction project.
- e. Publishes notices in the *Federal Register* of the successful completion of ITAAC in accordance with 10 CFR 52.99.
- f. Prepares an annual report on fabrication, manufacturing and construction of advanced power reactors and supports the Agency Action Review Meeting (AARM).

05.06 Division of Reactor Oversight (DRO, NRR)

- a. Provides the appropriate VIP staff to participate in inspections of advanced power reactors as necessary and in coordination with APO, Region II, and the host regions.
- b. Provides any significant quality insights as needed to ensure regulatory responses are appropriate.
- c. Coordinates with APO to develop and revise, as necessary, the VIP as it relates to traditional vendors for advanced power reactors fabrication, construction and manufacturing.
- d. Provides APO the status of planned VIP inspections related to advanced power reactors, including inspections related to ITAAC.
- e. Ensures that applicable inspection records are maintained in AIMS.

05.07 Regional Branch Chief(s) (DORS)

- a. Provides any significant quality insights as needed to ensure regulatory responses are appropriate.
- b. Implements project-specific inspection plans.
- c. Provides APO with the status of planned ARCOP inspections. This should be primarily accomplished through maintaining AIMS inspection records current.
- d. Ensure that applicable inspection records are maintained in AIMS.
- e. For Region 2, leads the ARCOP inspections in the “Quality of Reactor Plant Construction” strategic performance area.

2573-06 REQUIREMENTS

06.01 Inspection Planning. Project-specific inspection scoping matrices (project-specific matrices) shall be developed for each advanced power reactor construction project

06.02 Construction Inspection. The NRC staff will perform the inspections specified in the project-specific matrix to evaluate the licensee's safety-significant construction activities.

06.03 Inspection Results. The NRC staff will maintain the status of each reactor plant inspection plan using its project-specific matrix. The staff may modify the project-specific matrix using the guidance in this IMC.

2573-07 GUIDANCE

07.01 General

Advanced power reactors may be licensed under 10 CFR Part 50 or 52. When licensing a power reactor using 10 CFR Part 52, the Commission is required by 10 CFR 52.97(b)(1) to identify within the combined license the inspections, tests, and analyses

that the licensee shall perform and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act (as amended), and the Commission's rules and regulations.

ARCOP inspections employ vertical-slice inspection methodology, including inspections involving ITAAC. A vertical-slice inspection is an in-depth review of QA program attributes associated with the manufacture or construction of an SSC. This strategy not only verifies the quality of the SSCs inspected but provides confidence that other SSCs in the same inspection area will also be constructed with quality in accordance with the QA program. For example, a vertical slice inspection of safety-significant piping may include the QA program attributes of material procurement and control, design verification and control, welding, and testing. Verification that QA requirements have been met for these attributes would give the NRC confidence that the inspected piping will perform its safety function. Equally important, it would provide reasonable assurance that other safety-significant piping not directly inspected by the NRC would be fabricated and installed correctly and would be capable of performing its safety functions. During these inspections, inspectors should also evaluate the acceptability of ITAAC-related processes and conduct performance-based inspections of ITAAC completion, if applicable.

Successful completion of the ITAAC is the responsibility of the licensee; however, the licensee may delegate performance of the ITAAC to other entities (e.g., a manufacturer or project vendor). Licensees identify the bases for ITAAC completion in ITAAC Closure Notifications (ICNs). NRC staff review ICNs and ITAAC inspection results (when applicable) to verify that ITAAC are met.

10 CFR 52.103(g) requires the Commission to find that all acceptance criteria in the combined license are met prior to allowing the facility to operate. The Commission, in the Staff Requirements Memorandum (SRM) for SECY-13-0033, delegated the responsibility for the 10 CFR 52.103(g) finding to the NRC staff.

When an advanced power reactor plant is licensed under 10 CFR Part 50, requirements for ITAAC and a Commission finding pursuant to 10 CFR 52.103(g) are not applicable. Instead, the staff performs sufficient inspection to determine, pursuant to 10 CFR 50.57(a)(1), that construction of the facility has been substantially completed and conforms with the construction permit and the current operating license application, the provisions of the Atomic Energy Act (as amended), and the rules and regulations of the Commission to support a recommendation for issuing an operating license.

## 07.02 Design-Specific Matrix

The APO develops design-specific matrices in accordance with Attachment 2 of this IMC for each advanced power reactor design when the staff anticipates receiving an application for a CP, LWA, COL, or ML. Staff finalize the design-specific matrix following approval of the design and maintain the design-specific matrix current based on approved changes to the design. Changes to an approved design requested by individual licensees via license amendment or exemption requests should not normally be used to modify the design-specific matrix since the changes apply only to one licensee. These changes should be incorporated into the project-specific matrices for the appropriate reactor construction project.

## 07.03 Project-Specific Matrix

APO will coordinate with the construction inspection staff, DRO, and host regions as necessary to develop initial project-specific matrices in accordance with Attachment 2 of this IMC. Multiple project-specific matrices may be created for a single reactor construction project based on the number of units or reactors that are associated with the project, or other considerations. Reactors or units may also be combined into one or more matrices. APO will coordinate with Region II, DRO, and the host regions as necessary to maintain the project-specific matrices up to date in response to approved license amendments and exemptions, design changes, or modifications implemented under the 10 CFR 50.59 process or the change process in section VIII of the applicable design certification rule.

## 07.04 Nth-of-a-Kind (NOAK) Inspection Scoping

### a. Reactor Plant Sites (i.e., the site where the reactor will eventually operate)

The project-specific matrix used for a reactor plant being constructed or installed at the site where the reactor will eventually operate (excluding limited operation for testing at a manufacturing facility or other off-site facility) is the SSC baseline inspection scope for that reactor plant. The first baseline inspection scope for the first-of-a-kind (FOAK) assembled or constructed reactor plant is derived directly from the design-specific matrix. Subsequent reactor plants of the same design are referred to as nth-of-a-kind (NOAK) reactor plants. The baseline inspection scopes for NOAK reactor plants should be adjusted based on experience gained during oversight of the earlier reactor plants assembled or constructed.

Adjustments to the NOAK baseline inspection scopes are made by changing the project-specific matrix for NOAK reactor plants. These changes may include, but are not limited to:

- Regrouping SSCs in the project-specific matrix rows
- Redefining project-specific matrix inspection areas (matrix columns)
- Altering the minimum and maximum inspection samples per inspection area
- Altering the level of effort (inspection hours) associated with inspection samples

Each reactor plant's project-specific inspection scope includes sufficient inspection, when paired with confidence in quality gained from oversight of construction and assembly of previous reactor plants by the same entity, to determine that the reactor plant is built in accordance with its licensing basis.

### b. Manufacturing and Project Vendor Facilities

The initial project-specific matrix for reactor plants or significant portions of reactor modules being mass produced at a manufacturing facility or project vendor is the first-of-a-kind (FOAK) SSC baseline inspection scope. Subsequent SSC inspection scopes for NOAK reactors or reactor modules will be adjusted as described above. When a manufacturer or project vendor consistently demonstrates the ability to meet quality requirements for NOAK reactors or reactor modules, then the manufacturer or project vendor will transition to a periodic inspection plan. Periodic inspection plans are approved by the Director, APO and will be developed for the manufacturing or project

vendor facilities by APO staff. In developing periodic inspection plans, APO staff will consider the following:

- the design-specific scoping matrix
- the FOAK and subsequent project-specific scoping matrices
- the number of reactors or reactor modules previously assembled
- the number of reactors or reactor modules being assembled and/or tested at the facility per year (i.e., the throughput of the facility)
- the complexity of the reactor design and fabrication techniques
- previous NRC inspection results
- previous NRC assessment results
- the status of the manufacturer's or project vendor's corrective action program

The periodicity of inspections will be project-specific based on the factors above and should not normally exceed one inspection per year.

#### 07.05 Inspection

The NRC staff uses inspection procedure (IP) 75001, "Inspection of Manufacturing and Construction Quality for Advanced Power Reactor Structures, Systems, and Components" and its attachments listed in Attachment 1 of this IMC. Inspectors should evaluate if the assembly and construction of advanced power reactor modules and/or reactor plants is being accomplished in accordance with the licensing basis and other applicable regulatory requirements. During each inspection, inspectors review aspects of the implementation of the quality assurance program (QAP) using a vertical-slice inspection technique as described in IP 75001. Specific inspection guidelines for each inspection area are provided in the attachments to IP 75001.

Lead inspectors may also use a few hours during a site visit to review overall construction activities to verify the status of planned and in-process construction work. The status of these activities is used to assess inspection readiness and resource planning. The scope of the activities includes conducting plant walkdowns, talking with field workers, attending status meetings, and reviewing construction schedules.

NRC inspectors should implement one or more of the following inspection techniques: (1) direct inspection of in-process work activities, (2) review of completed records of work activities, or (3) independent assessment or inspection of completed work activities. The NRC will focus on direct inspections of assembly and construction activities when practical. When direct observation of assembly and construction activities is not practical, the SSCs may be inspected via records review and observation of the as-built condition of the SSCs.

The Attachment 1 IPs may be performed separately and should be performed at the appropriate time during assembly and construction for the items being inspected. However, to gain efficiency, 2 or more IP attachments may be performed concurrently. This IMC is intended to provide inspection requirements and guidance applicable to a wide variety of potential advanced reactor construction projects. These projects may vary greatly in scope, complexity, and risk to public health and safety. As a result, not all Attachment 1 IPs may be applicable or implemented at a specific facility. Applicable IPs will be identified in project-specific matrices through identification of inspection areas.

## 07.06 Inspection Planning

The staff develops and maintains a project-specific matrix (or matrices) for each reactor project. The project-specific matrices should be adjusted as necessary to account for prior inspection results including the assessment results of IMC 2572, "Assessment of Advanced Reactor Construction Projects." The level of effort and scope of inspection should be commensurate with the risk posed by the advanced reactor design to public health and safety and the environment. The baseline construction inspection scope and hours for each advanced power reactor facility will be documented in the project-specific matrix.

The number of project-specific matrices used for a reactor construction project will vary depending on the number of reactors included in the project, the complexity of the project, or other factors specific to individual projects. For example, a reactor construction project that includes multiple reactors may have a project-specific matrix for each reactor, or a single project-specific matrix that includes all reactors of the reactor construction project. The decision to use one or more project-specific matrices should be tailored to each project to maximize inspection efficiency.

## 07.07 Inspection Sample Selection

Inspections should be pre-planned based on expected SSC status at the time of inspection. The project-specific matrix contains information to aid inspectors in choosing the most appropriate inspection samples. This information includes operational and design/construction risk characterizations for each SSC, or group of SSCs in the matrix. Other project-specific factors should also be considered as appropriate.

In general, higher risk inspection opportunities should be prioritized over lower risk inspection opportunities. However, when SSC status is not as expected during an inspection, inspectors should use the project-specific matrix to choose any alternate available inspection opportunities in the matrix to inspect.

## 07.08 Inspection Results

Inspection results will be documented in accordance with IMC 0618, "Advanced Power Reactor Construction Inspection Reports." NRC staff should use AIMS to link inspection results to the appropriate docket, inspection report, and, when applicable, the specific ITAAC.

## 07.09 Assessment of Inspection Results

NRC staff will perform continual reviews of inspection results to determine if there is reasonable assurance of construction area quality. This assessment process is described in IMC 2572.

## 07.10 Enforcement

NRC staff will disposition ARCP findings in accordance with IMC 2571, "Dispositioning Advanced Power Reactor Construction Noncompliances" and the Commission's Enforcement Policy, NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions."

2573-08        REFERENCES

IMC 0618, "Advanced Power Reactor Construction Inspection Reports."

IMC 2507, "Vendor Inspections"

IMC 2570, "Advanced Reactor Construction Oversight Program General Guidance and Basis Document"

IMC 2571, "Dispositioning Advanced Power Reactor Construction Noncompliances"

IMC 2572, "Assessment of Advanced Reactor Construction Projects"

IMC 2574, "Inspection of the Advanced Power Reactor Operational Readiness, and Security and Safeguards Strategic Performance Areas"

NRC Enforcement Policy

END

Attachment 1: IMC 2573 Inspection Procedures

Attachment 2: The Design-Specific Inspection Scoping Matrix

Attachment 3: Revision History for IMC 2573

Attachment 1: IMC 2573 Inspection Procedures

Inspection procedure (IP) 75001, "Inspection of Manufacturing and Construction Quality for Advanced Power Reactor Structures, Systems, and Components."

IP or Attachment Number	IP Title
75001	Inspection of Advanced Reactor Construction Quality
75001.01	Foundations, Structural Steel & Structural Concrete
75001.02	Containment Integrity & Penetrations
75001.03	Piping, Pipe Supports & Restraints
75001.04	Reactor & Reactor Internals
75001.05	Mechanical Systems & Components
75001.06	Electrical Systems & Components
75001.07	Instrumentation and Control Systems & Components
75001.08	Ventilation and Confinement Systems & Components
75001.09	Fuel & Load Handling Systems
75001.10	Testing (ITP & PITAP)

## Attachment 2: Design-Specific Inspection Scoping Matrix

### Background:

The design-specific inspection scoping matrix (design-specific matrix) organizes structures, systems, and components (SSCs), and inspections, tests, analyses, and acceptance criteria (ITAAC), into groups that are closely related called “inspection areas.” This facilitates assessment of the quality of the inspection area based on inspection of representative SSCs within the same inspection area. Inspection areas are represented in the design-specific matrix as matrix columns. SSC inspection, along with inspection of the quality assurance processes in each inspection area, cumulatively provide the information needed to assess performance in each inspection area in accordance with IMC 2572.

The design-specific matrix provides a risk-informed, performance-based sampling approach to scoping ARCOP SSC inspections. The process for developing a design-specific matrix is technology-inclusive and scalable. The inspection areas of a design-specific matrix can be modified to support any advanced power reactor design, including large light water reactors. Each design-specific matrix is unique to a reactor plant design.

The design-specific matrix identifies the safety-related (SR) and non-safety related, safety-significant (NSRSS) SSCs included in the reactor plant design. For designs using the Licensing Modernization Program (LMP) methodology described in NEI 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” NSRSS SSCs are called non-safety related with special treatment, or NSRST. Therefore, the LMP design-specific matrix identifies the design’s SR and NSRST SSCs. For reactor designs that do not use the LMP methodology, NSRSS SSCs may be named differently. For example, the term “regulatory treatment of nonsafety systems,” or RTNSS, may be used. If other SSC classification schemes are used for reactor licensing, then the design-specific matrix should identify those SSCs that are SR and NSRSS. These SSCs occupy the matrix rows of the design-specific matrix.

SSCs related to operational, security, and safeguards programs, such as fire protection, emergency planning, security, and radiation protection, are not included in the design-specific matrix. Instead, they are scoped for inspection using IMC 2574, “Inspection of the Advanced Power Reactor Operational Readiness Strategic Performance Areas,” and IMC 2203, “Security Inspection Program for Advanced Power Reactor Construction.” These program SSC inspections may be included in project-specific matrices for convenience or may be tracked separately as part of the program inspections described in IMC 2574. If included in a project-specific matrix, the program SSCs should not be comingled with SR and NSRSS SSCs. Instead, they should be placed in a separate inspection area created for this purpose.

The design-specific matrix identifies the principal inspection procedure applicable to each inspection area; identifies ITAAC associated with the SSCs, if applicable; provides SSC operational and design/construction risk insights to risk-inform inspections; specifies the minimum and maximum inspection sample sizes for each inspection area; and identifies the fundamental safety function (FSF) each SSC supports.

The design-specific matrix is converted into a project-specific inspection scoping matrix (project-specific matrix) where project-specific information is used to modify the design-specific matrix. As inspections are completed, inspection samples are recorded in the project-specific matrix in the ARCOP Information Management System (AIMS). The project-specific matrix provides NRC staff and management a means to assess baseline inspection program progress and results.

### Developing The Design-Specific and Project-Specific Matrices:

The APO will follow guidelines below to develop design-specific matrices for each advanced power reactor design when the staff anticipates receiving an application for a construction permit (CP), limited work authorization (LWA), combined license (COL), or manufacturing license (ML). The staff may begin development of the design-specific matrix based on design information provided in the application for a standard design approval (SDA), a design certification (DC), or custom design. Once established, the design-specific matrix may be used to create any number of project-specific matrices. Licensees may have additional insights that will aid the NRC in developing the design and project matrices. The staff should engage with licensees and share matrix draft copies at appropriate points in the matrix development process to gain these insights.

1. NRC personnel with some combination of expertise in reactor plant design, construction, inspection, and risk assessment should populate the design-specific matrices. Consider adding additional technical experts, including contractors, with expertise in the new design technologies, materials, and construction techniques, if necessary.
2. For each design, create a design-specific matrix using the generic Design-Specific Inspection Scoping Matrix Template (Figure 1). APO should revise the generic template as appropriate as program experience is gained. Modify the generic template as necessary to support the design being reviewed. This may include adding new inspection areas (i.e., columns) and IPs, deleting inspection areas that are not applicable to the design, or combining inspection areas for efficiency.
3. Based on review of the reactor design, populate the design-specific matrix rows with SSCs that perform or directly support an FSF. This includes SR and NSRSS SSCs. However, not all SR and NSRSS SSCs must be included in the matrix. Staff may omit SSCs from the matrix that have little risk-significance and inspection value. Other non-safety related (NSR) SSCs that support operational programs, such as emergency planning, fire protection, security, radiation protection, and effluent monitoring should not be included in the design matrix. Instead, these SSCs should be scoped for inspection using IMC 2574 or IMC 2203 and may be included in the project-specific matrix.

As the matrix rows are populated with the appropriate SSCs, the SSCs may be grouped into a more general collection of similar SSCs to facilitate efficient inspection of those SSCs. For example, instead of listing every mechanical SSC in a reactor building ventilation system, such as duct work, isolation dampers, etc., those SSCs may be grouped together in one row labeled "Reactor Building Ventilation System Mechanical Components."

- a. For applications that use the LMP methodology, use the following guidance when selecting SSCs to include in the design matrix:

- Safety-Related SSCs:

SSCs selected by the designer from the SSCs that are available to perform the required safety functions (RSFs) to mitigate the consequences of design base events (DBEs) to within the licensing basis event frequency-consequence (LBE F-C) Target, and to mitigate design basis accidents (DBAs) that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.

SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of beyond design basis events (BDBE) with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target.

- Non-Safety Related with Special Treatment (NSRST):

Non-safety-related SSCs relied on to perform safety-significant functions. Safety-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C Target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.

In addition, non-safety related SSCs relied on to provide adequate Defense-in-Depth (DID) are classified as NSRST. NSRST SSCs for DID purposes can be included in the matrix to provide maximum flexibility in implementing the program. However, these SSCs should only be selected for inspection if inspection of their special treatment would provide meaningful insights into QAP implementation for risk-significant SSCs.

- Non-Safety-Related with No Special Treatment (NST):

All other NSR SSCs (with no special treatment required) are not safety-significant and should not be entered into the matrix.

b. For applications that do not use the LMP methodology, use the following guidance when selecting SSCs:

- Safety-Related SSCs:

Safety-related structures, systems and components are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or §100.11 of Part 50, as applicable.

- Non-safety related, safety-significant (NSRSS) SSCs:

NSRSS SSCs may be referred to as SSCs "important to safety." See NUREG 0800, Standard Review Plan for criteria of NSRSS SSCs. NSRSS SSCs can be included in the matrix to provide maximum flexibility in implementing the program. However, these SSCs should only be selected for inspection if inspection would provide meaningful insights into QAP implementation for risk-significant SSCs.

- All other Non-Safety-Related SSCs:

All other NSR SSCs are not safety-significant and should not be entered into the matrix.

For each SSC, or group of SSCs, selected into the matrix rows, enter the system or component classification, and associated FSF(s).

4. For each SSC, determine a rating (High (H), Medium (M), or Low (L)) for Risk Importance - Design ( $RI_D$ ) and Risk Importance - Construction ( $RI_C$ ). Provide a basis, when appropriate, for the classification assigned. Avoid classifying everything as either high or low risk. These classifications are used by inspectors to risk-inform their inspections in each inspection area. Use available risk information, knowledge of the design, engineering judgement, and construction/manufacturing experience and the following guidance:
  - a. Risk Importance - Design ( $RI_D$ ): Provides a focus on the most important activities or SSCs for public safety.  $RI_D$  is based on the available risk information. Probabilistic risk assessments (PRAs), or other risk analyses, may be used to inform the  $RI_D$ , if available. Assign  $RI_D$  for each matrix row based on the relative importance or significance of the SSCs in fulfilling fundamental safety functions. Enter the  $RI_D$  (H, M, or L), into the  $RI_D$  column of the matrix.
  - b. Risk Importance - Construction ( $RI_C$ ):  $RI_C$  includes consideration of: (1) the complexity of the SSC design, (2) the amount of industry experience fabricating or constructing the SSC (e.g., is the SSC first of a kind?), (3) the complexity of the fabrication or construction process, (4) other available information about construction experience associated with the SSC, and (5) the opportunity to verify SSC quality by other means (such as pre-operational testing). When assigning  $RI_C$ , NRC staff should consider each of these attributes and provide a basis to explain which attribute(s) significantly contributed to the final  $RI_C$ . Enter the  $RI_C$  (H, M, or L), into the  $RI_C$  column of the matrix.
5. For each matrix row, identify the applicable inspection area(s) for the SSC(s) and place an "X" in the appropriate matrix cells.
6. Optional: For matrix rows, the anticipated location of inspection may be entered.
7. The matrices created for each advanced reactor design, and reactor construction project, are living documents that are updated based on lessons learned, updated design information, and design changes. Additionally, as the NRC staff gains experience inspecting a particular design, they should use the prior inspection history, lessons learned, prior ARCP assessment results, and other available information to modify the scope of the design-specific matrices. NRC staff also incorporate this information into the individual project-specific matrices for sites or facilities fabricating, manufacturing, or constructing the same reactor design, and for a project or manufacturer's NOAK reactors.
8. Site-specific SSCs and site-specific ITAAC are used to augment the design-specific matrix when developing the project-specific matrix.
9. For each inspection area, specify the minimum and maximum number of inspection samples and enter the sample numbers into the matrix.
  - a. Minimum sample sizes are based on the minimum expected inspection samples necessary to provide reasonable assurance that SSCs in an inspection area are being fabricated/manufactured/constructed with adequate quality and meet their licensing basis functional and qualification requirements.

- b. Maximum sample sizes are based on the maximum expected inspection samples necessary to provide reasonable assurance that SSCs in an inspection area are being fabricated/manufactured/constructed with adequate quality and meet their licensing basis functional and qualification requirements. Maximum sample sizes may only be exceeded with regional division director approval and with concurrence from the director of APO. Circumstances requiring inspection activity beyond the maximum specified in the project-specific matrix may include:
- 1) the maximum number of inspection samples did not include inspection of quality assurance processes vital to a reasonable assurance determination, or
  - 2) NRC findings were identified that prevent a reasonable assurance determination in an inspection area, or
  - 3) additional information has become available that brings into question a previous reasonable assurance determination in an inspection area, or
  - 4) quality assurance processes and/or organizations have been changed such that inspection of the new processes or organizations is required to verify a previous reasonable assurance determination.

Management Review:

The Director, APO shall approve each design matrix.

Figure 1: Generic Design Specific Inspection Scoping Matrix

SSC(s)	Fundamental Safety Function <sup>1</sup>	RI <sub>D</sub>		RI <sub>C</sub>		Reactor Plant System	Safety Class <sup>2</sup>	Individual Columns for each Inspection Area applicable to the Adv. Reactor Design <sup>3</sup> .					Location Factor (F) Site (S) Both (FS)
	(RC) (HR) (RR)	(H)(M)(L)	Basis	(H)(M)(L)	Basis								
INSPECTION PROCEDURES <sup>4</sup> 75001 and 75001.XX													
MINIMUM INSPECTION SAMPLES <sup>5</sup>													
MAXIMUM INSPECTION SAMPLES <sup>5</sup>													

Notes:

1. The fundamental safety functions (FSFs) are defined as Controlling Heat Generation (reactivity and power control) (RC), Controlling Heat Removal (including reactor and spent fuel decay heat generated from waste stores) (HR), and Retaining Radionuclides (RR).
2. Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST), Regulatory Treatment of Non-Safety-Related SSCs (RTNSS), or other design-specific classification terminology. Also include if the SSC is risk-significant (RS) or defense in depth (DID) when available.
3. The specific Inspection Area (i.e., matrix column) and its associated IP along with additional supporting IPs are listed in Attachment 1 to this IMC. Not every inspection area and IP will be applicable to every advanced power reactor design.
4. The general guidance in IP 75001 is applicable to every inspection and is only listed once. The specific IP attachment (i.e., 75001.XX) is listed in the applicable Inspection Area column. Note that more than one inspection area and IP are commonly applicable to an SSC.
5. The minimum and maximum inspection samples constitute the minimum and maximum baseline inspection program for a given design and are unique to each design specific matrix.

### Attachment 3: Response to Significant Issues or Events

This attachment provides guidance for responding to significant events at advanced power reactor construction, project vendor, or manufacturing sites subject to the advanced reactor construction oversight program (ARCP). A significant event is a radiological, safeguards, security, or other event that poses an actual or potential hazard to public health and safety, common defense and security, property, or the environment (ref. Management Directive (MD) 8.3, "NRC Incident Investigation Program").

The NRC uses deterministic criteria to decide if a reactive inspection (Special Inspection Team (SIT) or Augmented Inspection Team (AIT)) should be conducted at a manufacturing, project vendor, or construction site. Since irradiated fuel is typically not present at these sites, there is no radiological threat to the public. Therefore, an Incident Investigation Team (IIT) should not be appropriate for events at these sites. MD 8.3 defines the authorities, responsibilities, and basic requirements of personnel investigating significant events.

Upon notification of a significant event, the cognizant inspection organization (Region II construction inspection organization for on-site construction events, the host region for manufacturer, project vendor, operational readiness, or security events) staff should perform the initial review to assess the significance of the event and the level of response required. The cognizant inspection organization staff should also ensure that APO staff are promptly notified of the event. APO staff will assist the cognizant inspection organization as necessary.

If the event meets one or more criteria for an SIT listed below, the host region Regional Administrator (RA), in consultation with the Director, APO and the Director of the cognizant inspection division, makes the decision to initiate an SI or not. The cognizant inspection division will recommend to the host region RA the SIT leader and members.

If the event meets one or more of the AIT criteria, APO staff will coordinate with the appropriate technical branches to provide input into the decision to dispatch an AIT or not. If the event has security-related aspects, then the Division of Preparedness and Response (DPR) in NSIR should also be consulted. The host region RA consults with the Director of NRR and, if applicable, the Director of NSIR to decide on the appropriate response. Per MD 8.3, if consensus cannot be attained, then the Executive Director of Operations (EDO) makes this decision.

In addition to criteria listed in MD 8.3, the following are additional criteria considered in evaluating what type of response, if any, is appropriate:

- a. Any significant weather-related event or natural disaster (hurricanes/tornados, earthquakes, fire, flooding, etc.) that may have significant impact on structures, systems, and components (SSCs), or that may invalidate completed ITAAC. The staff should consider the use of either an SIT or AIT depending on the type and amount of damage the facility sustained. The purpose of either would be to monitor and assess the licensee's actions to recover damaged or potentially damaged safety-significant SSC's. If the event involves the loss or damage of special nuclear material (SNM) or sources, coordination with state and local governments should be part of the response and should generally result in an AIT.
- b. Any significant security-related issue (e.g., loss or theft of SNM, or potential tampering/sabotage). Either an SIT or AIT should be considered depending on the complexity and significance of the issue. Issues such as the loss or theft of SNM or

confirmed tampering or sabotage should generally result in an AIT. Issues such as potential tampering, multiple FFD issues, or an unauthorized, actual discharge of a weapon should generally result in an SIT.

- c. Onsite accidents resulting in significant damage to safety-significant SSCs or invalidating ITAAC (e.g., crane collapse, train or other significant vehicle accident). Consideration of either an SIT or AIT is appropriate depending on the type and amount of damage sustained. As with responses to weather events discussed above the purpose of either would be to monitor the licensee's recovery from damaged safety-significant SSCs or SSCs related to ITAAC.
- d. Significant offsite or onsite industrial events that impact the site (e.g., hazardous chemical spill, nearby chemical plant or refinery fire, etc.). An SIT may be appropriate if there is a possibility of significant impact on constructed items or materials.

Consideration of whether airborne chemical fumes could have an adverse impact on risk significant SSCs or other program elements with ITAAC or material in storage should be given. For instance, chlorine gas that contacts stainless steel items may be detrimental. The purpose of the inspection would be to ensure that the licensee has conducted an adequate evaluation of any potential impacts, including extent of condition. Generally, an AIT would not be warranted.

- e. Stop work order issued by the licensee for which the underlying quality issue(s) are not already fully understood. The use of an SIT may be appropriate to ensure that the NRC fully understands the underlying issues. Generally, an AIT would not be warranted.
- f. Labor dispute. The use of an SIT may be appropriate to review and/or monitor licensee actions to ensure that malicious mischief is not taking place that could impact the quality of construction. Generally, an AIT would not be warranted.
- g. Potential financial impact on programs or quality of work. A review of the licensee's quality oversight of construction activities with an SIT may be appropriate to determine if degradation of quality or programs is occurring. Inspection or review of the licensee's finances is not appropriate. Generally, an AIT would not be warranted.
- h. Significant safety conscious work environment (SCWE) issues or allegations which do not have a specific performance aspect that could be addressed thorough the IMC 2572 assessment process or independent licensee action. The use of an SIT may be appropriate. Generally, an AIT would not be warranted.
- i. Any significant issue not covered by the above that NRC management judges to warrant additional inspection beyond what can be accomplished through adjustments to the baseline inspection scope. The use of an SIT may be appropriate. The use of an SIT is generally not appropriate for quality assurance breakdowns. Instead, the use of focused baseline inspections should be considered for these issues.

If an advanced reactor under construction is co-located with an operating facility, NRC oversight organizations should coordinate the response if the event affects both the reactor under construction and the operating reactor. Coordination is important to ensure that any response to an event at a construction site does not have an adverse impact on the operating site.

Inspectors responding to an event at a construction site should be sensitive to potential impacts

to the operating facility and promptly communicate those to the NRC operations oversight staff for the facility.

Exhibit 1 provides a form for cognizant inspection organization staff to use when documenting the decision to perform a reactive inspection based on evaluation of the criteria listed in MD 8.3 and above. As noted in Exhibit 1, cognizant inspection organizations may customize the form to fit specific organizational protocols, but the criteria should not be changed.

- a. If none of the deterministic criteria were met, briefly document the key points of discussion in the "Remarks" section that were the principal focus areas of the evaluation. Also, state that no criteria were met in the "Response Decision" section of the form.
- b. If one or more of the criteria were met, briefly indicate the basis for each in the "Remarks" section of the applicable criteria.
- c. Use the "Response Decision" section to provide the basis for deciding whether to conduct an inspection, and which level of inspection is recommended. Maintain a record of the decision by placing the evaluation results in Agencywide Documents Access and Management System (ADAMS).
- d. Whenever an SIT or AIT is planned, the host region should also notify the licensee of its intentions once a final decision is made.
- e. NRC staff planning and conducting the SIT or AIT should follow the process outlined in MD 8.3.

IMC 2574, Exhibit 1: Decision Documentation for a Construction

SI/AIT

SITE:	EVENT/ ISSUE DATE:	EVALUATION DATE:
-------	--------------------	------------------

Brief Description of the Event/Issue:

Significant Weather-Related, Natural Disaster, or Man-Made Event

Y/N	<u>SI Deterministic Criteria</u>
	Significant damage to safety-significant SSCs or SSCs related to ITAAC.
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Extensive damage to safety-significant SSCs or SSCs related to ITAAC.
	Remarks:
	Involved a loss or damage to SNM
	Remarks:
	Involved a loss or damage to radiological sources resulting in dose to an individual in excess of applicable regulatory limits.
	Remarks:

Significant Security-Related Event

Y/N	<u>SI Deterministic Criteria</u>
	Potential tampering or sabotage
	Remarks:
	Unauthorized, actual discharge of a weapon
	Remarks:
	Multiple FFD issues
	Remarks:
	Other (explain in remarks)
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Loss or theft of SNM
	Remarks:
	Confirmed tampering or sabotage
	Remarks:
	Other (explain in remarks)
	Remarks:

<u>Onsite Event Resulting in Significant Damage to Safety-Significant SSCs</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Significant damage to safety-significant SSCs or SSCs related to ITAAC.
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Extensive damage to safety-significant SSCs or SSCs related to ITAAC.
	Remarks:
<u>Significant Offsite or Onsite Industrial Event</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Possibility of significant impact on stored or constructed items or materials.
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/if "yes," then provide rationale in remarks block.
	Remarks:
<u>Stop Work Order Issued by the Licensee or Permit Holder</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Stop work order for which the underlying issues are not fully understood.
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/ if "yes," then provide rationale in remarks block.
	Remarks:
<u>Labor Dispute</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Labor Dispute
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/ if "yes," then provide rationale in remarks block
	Remarks:
<u>Potential Financial Impact on Programs/Quality</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Potential financial impact on programs or quality
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/ if "yes," then provide rationale in response decision block
	Remarks:

<u>Significant SCWE Issue or Allegation</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Significant SCWE issue or allegation that cannot be addressed through IMC 2572 or independent licensee action
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/ if "yes," then provide rationale in remarks block.
	Remarks:
<u>Other Significant Event</u>	
Y/N	<u>SI Deterministic Criteria</u>
	Other criteria listed in MD 8.3 or a significant issue not covered and judged by management to warrant additional inspection or follow up.
	Remarks:
Y/N	<u>AIT Deterministic Criteria</u>
	Other/ if "yes," then provide rationale in remarks block.
	Remarks:

<u>RESPONSE DECISION</u>	
Using the above information and other key elements of consideration as appropriate, document the response decision to the event or issue, and the basis for the decision.	
Decision and Details of the Basis for the Decision:	
Cognizant Inspection Division Review (Director or designated representative):	
Date:	
APO Review (Director or designated representative):	Date:
Host Region RA, or designated representative, Approval:	Date:

Note: The above tables are provided as examples only. Cognizant inspection organizations have discretion to modify these tables in their implementing procedures or office instructions.

Attachment 4: Revision History for IMC 2573

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	ML25210A581	Draft IMC for public comment	N/A	N/A
N/A	ML25342A171 02/05/26 CN 26-004	Initial Issuance.	Construction Inspector, supervisor and PM ARCOPE training	ML25336A292