United States Nuclear Regulatory Commission
DRAFT Action Plan for Advanced Manufacturing Technologies (AMTs)
Appendices

Appendix A – Examples of NRC Review of Prior Submittals

A.1 Peening of Welded Austenitic Stainless Steel Spent Fuel Storage Canisters

Laser peening of welded austenitic stainless-steel canister welds has been utilized to improve the resistance of the canister welds to chloride induced stress corrosion cracking (CISCC). Testing and analyses conducted by the storage system vendor showed that the process provided adequate residual stress mitigation of the canister welds. Analyses included the measurement of residual stress profiles including the depth of the imparted compressive stress by the laser peening process. Testing included a demonstration that the laser peened austenitic stainless steel welds were resistant to stress corrosion cracking in concentrated chloride solutions at elevated temperatures. The vendor also implemented the necessary process controls necessary to consistently apply laser peening to ensure adequate CISCC mitigation of each canister.

A.2 Metal Matrix Composite Neutron Absorbers

The basket in spent fuel dry storage and transportation systems maintains the internal geometry of the fuel. The basket typically includes neutron absorber materials for criticality control. Early storage systems used Boral™ which was a composite composed of aluminum or aluminum alloy cladding over a boron carbide core. Operational issues with Boral™ in spent fuel pools prompted the development of metal matrix composite (MMC) neutron absorbers constructed using aluminum and boron carbide with a powder metallurgy process. Testing of these fully dense composite materials have shown them to be robust over the range of conditions encountered during spent fuel loading and drying operations for dry storage and transportation systems. More recent developments with MMC neutron absorber materials include the use of nanoparticles to reinforce the matrix alloy. These materials have been used as both the neutron absorber material and the structural material for the basket.

Prior to use, applicants must provide information to qualify the neutron absorber material for criticality control. For neutron absorbers that are also structural materials, the applicant must provide mechanical properties over the range of operational temperatures and an analysis to show that the materials maintain criticality control under normal, off-normal and accident conditions.

A.3 Peening of Reactor Vessel Head Nozzles and Nickel-Based Dissimilar Metal Welds

Peening for use on nuclear grade piping and components uses a variety of specialized methods (e.g., water jet, cavitation, and laser) to put the wetted surface of a susceptible material into a compressive stress state. If properly applied, this compressive stress state will help prevent the initiation of new fatigue and stress corrosion cracking without damaging the original material. Peening can be performed under 10 CFR 50.59 without prior NRC approval.

Licensees desire to peen reactor coolant pressure boundary components: (1) for asset preservation, and, (2) to support requests for volumetric inspection relief.

Inspection relief for reactor vessel head nozzles and nickel-based dissimilar metal welds may be obtained through a plant-specific relief request. Licensees may reference the NRC approved topical report Electric Power Research Institute Material Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement, MRP-335, Revision 3-A (MRP-335 Revision 3-A) to support plant-specific relief requests.

NRC staff has approved requests for inspection relief from Byron and Braidwood Units 1 and 2 for peened reactor pressure vessel head penetration nozzles. The NRC is currently reviewing the first relief request volumetric inspection relief for dissimilar metal butt welds at Callaway.

A.4 High Density Polyethylene (HDPE) Pipe

At the time of the design and construction of the current USA nuclear fleet, the ASME Code Section III and Section XI did not have rules for material, design, fabrication, installation, and testing of high density polyethylene (HDPE) piping. However due to general corrosion as well as microbiologically influenced corrosion (MIC) occurring in service water carbon steel piping, nuclear operating plants explored the possibility of using HDPE piping. Several licensees submitted relief requests to replace certain portions of their low pressure safety-related ping with HDPE piping. The NRC staff reviewed these requests on their individual merits and approved each request (ADAMS Accession Nos. ML083100288, ML091240156, and ML15337A414).

As a result of industry interest to use HDPE piping and the benefits that HDPE piping offered for certain applications, the Office of Nuclear Reactor Regulation (NRR) requested that the Office of Nuclear Regulatory Research (RES) assist NRR and provide support for research efforts in the evaluation of HDPE fusion joining process and nondestructive examinations of HDPE piping and fittings for repair replacement and inspection activities related to its use by operating nuclear plants. These efforts are in part documented in following publications, “Progress on the Evaluation of Ultrasonic Phased-Array Examination for Fabrication Flaw Detection in HDPE Butt Fusion Joints,” November 2016 (ADAMS Accession No. 17003A106)., and “Summary of Research Projects at Emc^2 Conducted Under Contract to the US NRC” (ADAMS Accession No. ML18144A058).

Concurrently and also due to the above mentioned interest, ASME began working on Code Case N-755, which has since been incorporated into ASME Code Section III, 2015 edition as Mandatory Appendix XXVI, which addresses the use of HDPE piping in buried safety related to ASME Code Class 3 piping systems.

A.5 Carbon Fiber Reinforced Polymer Systems

Currently, there are no provisions in the ASME Code, Section XI, or in an approved ASME Code Case for installing carbon fiber reinforced polymer (CFRP) systems as a replacement for carbon steel piping during a repair/replacement activity. Two instances of the use of carbon fiber reinforced polymer systems are described below:

- Virginia Electric And Power Company (Dominion) submitted a proposed alternative the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWA-4000, for the repair of buried piping, pursuant to 10 CFR 50.55a(z)(1), by applying layers of CFRP to the inside surface of buried circulating water and service water piping at Surry Power Station (ADAMS Accession No. ML16355A346). The NRC staff’s review was completed in approximately one year and involved the licensee providing a supplement to its proposed alternative and responding to numerous Requests for Additional Information. Ultimately, the NRC staff reviewed and approved this proposed modified alternative on a plant-specific basis because the proposal provides reasonable assurance of structural integrity and leak tightness of the buried circulating water and service water piping (ADAMS Accession No. ML17303A068). The NRC staff only authorized the use of the modified alternative for the fifth and sixth 10-year ISI intervals at the Surry, Unit 1, and for the fifth 10-year ISI interval at Surry, Unit 2.

- Entergy Nuclear Operations, Inc. applied a nonstructural external coating of composite (carbon fiber epoxy) material on portions of the service water piping in the Zurn strainer pits, to protect the pipe from mechanical damage or loss of material (ADAMS Accession No. ML17250A244). The licensee also repaired a non-safety-related service water pipe by applying layers of CFRP on the outside surface. This repair was "designed to act as the original piping should the weld fail, and structural integrity [be] compromised" (ADAMS Accession No. ML17250A244). The NRC staff's review consisted of an onsite audit conducted on August 1-3, 2017 (ADAMS Accession No. ML17250A244) and applicant responses to the Requests for Additional Information and audit activities by letters dated December 14, 2017 (ADAMS Accession No. ML17360A158); December 21, 2017 (ADAMS Accession No. ML17363A213); and February 26, 2018 (ADAMS Accession No. ML18064A136). The applicant's responses resulted in the following changes and enhancements to its aging management programs. The NRC staff reviewed and approved the licensee's proposal to manage aging associated with CFRP wrap of these components during plant operation in the license renewal period (ADAMS Accession No. ML18200A333).

Appendix B – Applicable Regulatory Requirements

A brief synopsis of the applicable regulatory requirements follows. This summary highlights the most significant relevant regulatory requirements.

- 10 CFR Part 50.34, 52.47, 52.79, 52.137, and 52.157 require that an application for a construction permit, design certification, combined license, design approval, or manufacturing license must include principal design criteria for a proposed facility. These criteria are to provide the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SCC) important to safety. Pertinent required information includes documentation of compliance with 10 CFR Part 50 Appendices A and B.

- 10 CFR Part 50, Appendix A provides minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location for which construction permits were issued by the Commission during the drafting of 10 CFR Part 50 Appendix A. GDCs relevant to this action plan include 1, 2, 4, 10, 14, 15, 30, 31, 32, 36, and 37.

- 10 CFR Part 50 Appendix B provides quality assurance criteria for nuclear power plants including quality assurance program; design control; procurement document control; instructions, procedures, and drawings; document control; control of purchased material, equipment, and services; identification and control of materials, parts, and components; control of special processes; inspection; test control; control of measuring and test equipment; handling, storage, and shipping; inspection, test, and operating status; nonconforming materials, parts, or components; corrective action; quality assurance records; and audits.

- 10 CFR Part 50.55a incorporates by reference ASME Code Sections III and XI, along with Code Cases for these sections, with conditions. 10 CFR Part 50.55a also accepts ASME NQA-1, “Quality Assurance Requirements for Nuclear Facilities”. Compliance with 10 CFR 50.55a provides assurance for meeting the associated GDCs.

- 10 CFR Part 50.59 allows licensees to make changes to their facility, procedures, and/or conducts tests and experiments not described in their final safety analysis report (FSAR) without obtaining a license amendment if:
    - A change in the technical specification is not required,
    - The change, test, or experiment does not result in more than a minimal increase in frequency of occurrence of accident, likelihood of occurrence of malfunction of SSC important to safety, consequence of an accident, or consequences of a malfunction of an SSC important to safety than previously reported in the FSAR.

- 10 CFR Parts 50.60, 50.61, 50.61a, Appendix G, and Appendix H provide additional fracture toughness requirements pertaining to the reactor coolant pressure boundary.

- 10 CFR Part 50.69 provides voluntary alternative requirements pertaining to the above cited regulatory requirements based on risk-informed treatment of SSCs. The risk-informed requirements may be applied via license amendment. The categorization that is most closely staff-scrutinized is the designation of an SSC as a RISC-3, namely safety-related but performing low safety significant function.

- 10 CFR Part 54 governs the issuance of renewed operating licenses for nuclear power plants. Part 54 addresses, in part, the adequacy of applicant activities to manage the effects of aging for those systems, structures, and components that are within the scope of the rule and are passive ("perform an intended function without moving parts or without a change in configuration or properties") and long lived ("not subject to replacement based on a qualified life or specified time period"), for the period of extended operation.

- 10 CFR Part 72.24 identifies the minimum information that must be included in a Safety Analysis Report describing the proposed independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS) for the receipt, handling, packaging, and storage of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste as appropriate. This includes the design criteria for the ISFSI or MRS pursuant to 10 CFR Part 72 Subpart F of this part, with identification and justification for any additions to or departures from the general design criteria; the design bases and the relation of the design bases to the design criteria; information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety; and applicable codes and standards.

- 10 CFR Part 72.48 allows licensees or certificate of compliance holders to make changes to their facility or spent fuel storage cask design as described in the FSAR, and conduct tests or experiments not described in their FSAR, without obtaining a license or certificate amendment if:
    - A change in the technical specification is not required,
    - A change in the terms, conditions, or specifications incorporated into the certificate of compliance is not required;
    - The change, test, or experiment does not result in more than a minimal increase in frequency of occurrence of accident, likelihood of occurrence of malfunction of SSC important to safety, consequence of an accident, or consequences of a malfunction of an SSC important to safety than previously reported in the FSAR.

- 10 CFR Part 72.236 describes the requirements for spent fuel storage cask approval and fabrication. This includes content specifications, identification of design bases and design criteria for structures, systems, and components important to safety; specific performance requirements, and required design features and inspections.

- 10 CFR Part 72 Subpart G identifies the quality assurance program requirements for the design, fabrication, construction, testing, operation, modification, and decommissioning of the structures, systems, and components of the ISFSI or MRS important to safety.

- 10 CFR Part 71.33 identifies the required radioactive material transportation package description including identification of packaging classification, materials of construction and fabrication methods, and package contents

- 10 CFR Part 71.35 identifies the required transportation package evaluations including package approval standards in 10 CFR 71 Subpart E and tests in 10 CFR 71 Subpart F.

- 10 CFR Part 71 Subpart H identifies the quality assurance requirements for radioactive material transportation packaging applying to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety.

- 10 CFR Part 70.22 describes the contents of applications for a license for a proposed fuel fabrication or plutonium processing facility.

- 10 CFR Part 70.64 provides the requirements for new facilities or new processes at existing facilities.

- 10 CFR Part 70.72 describes the facility change process and allows licensees to make changes to their facility and procedures without prior NRC approval if certain criteria are met.

Appendix C – EXTERNAL ORGANIZATIONS

C.1 Codes and Standards Organizations

The NRC has active involvement in many of the Codes and standards (C&S) organizations that could develop C&S for AMT, including ASME and ASTM.

American Society of Mechanical Engineers Boiler and Pressure Vessel Code

The NRC routinely participates in the activities of the ASME Boiler and Pressure Vessel Code committees, with a heavy emphasis in the committees responsible for Section III (Rules for Construction of Nuclear Facility Components) and Section XI (Rules for Inservice Inspection of Nuclear Power Plant Components) of the Code. These interactions enable the staff to readily assess new editions of the ASME Code to determine their acceptability for incorporation by reference in 10 CFR 50.55a.

Specific to AMT, the ASME Board on Pressure Technology Codes and Standards and the ASME Board on Nuclear Codes and Standards have formed a joint BPTCS/BNCS Special Committee on Use of Additive Manufacturing for Pressure Equipment. The charter of this groups is:

> *To develop a technical baseline to support development of a proposed BPTCS standard or guideline addressing the pressure integrity governing the construction of pressure retaining equipment by additive manufacturing processes. Construction, as used in this Charter, is limited to materials, design, fabrication, examination, inspection, and testing.*

The NRC has a representative on this special committee.

For a typical introduction of new materials, processing, analysis, or inspection requirements desired by an entity affected by the Code (manufacturer, plant operator, etc.), the appropriate ASME task or working group develops a Code Case to approve alternatives to ASME Code provisions, for such topics as materials, fabrication, and inspection aspects. After some time period of implementation, the provisions of the Code case, if found acceptable, may be introduced into the Code itself. The NRC reviews the Code cases approved by ASME Sections III and XI and makes a determination on their acceptability in Regulatory Guides 1.84 (Section III) and 1.147 (Section XI). As stated previously, the NRC assesses each edition of the ASME Code and considers incorporation by reference in 10 CFR 50.55a.

ASME Verification and Validation in Computational Modeling and Simulation

The NRC routinely participates in the activities of the ASME Verification and Validation committees. The charter of this groups is to coordinate, promote, and foster the development of standards that provide procedures for assessing and quantifying the accuracy and credibility of computational models and simulations. One of the subcommittees of this organization is focused specifically on computational modeling for advanced manufacturing. The NRC has a representative on the main V&V committee who routinely interacts with this subcommittee.

ASTM International

ASTM International, formerly the American Society of Testing and Materials, formed the F-42 committee "Additive Manufacturing Technologies" in 2009. This ASTM committee is responsible for writing standards for materials and processes used in AM. The NRC has participants in the activities of this committee.

Other C&S Groups

For other C&S organization (NACE, ANS, etc.), the NRC remains cognizant of their AMT standardization activities. In part this is achieved by following the activities of organizations such as the America Makes & ANSI Additive Manufacturing Standardization Collaborative as they develop standardization roadmaps.

NRC staff is actively encouraging NACE to become involved in writing standards for corrosion properties of AM metal components. When available, the NRC will engage with the appropriate NACE committees.

C.2 Other Government Organizations

Other government organizations have been very active in the development, standardization, use, and regulation of AMTs. The NRC will review and build on those foundations as applicable to its mission related to the nuclear industry.

Regulatory Organizations

The NRC will engage with other government regulatory agencies to increase knowledge on the technical aspects of AMT as well as the regulations developed for the acceptance of products made using AMT. The Federal Aviation Administration (FAA) has developed a regulatory framework to guide the creation of regulations for the use of additive manufacturing (AM) in aviation applications. The NRC attended a workshop, hosted by the FAA, which discussed the progress of regulations and presented research and development performed by aviation companies. The NRC will continue to monitor the progress of the FAA and utilize any common information to help develop stronger regulations.

Non-Regulatory Organizations

The NRC will leverage NASA standards development activities associated with additive manufacturing technologies for building spacecraft applications requiring high reliability. NASA has begun to establish consensus standards defining requirements of AM process parameters (e.g. powder controls, process parameters, chamber environment, materials integrity, and mechanical properties). NRC intends to engage NASA to better understand knowledge gaps associated with AM process parameters. NASA's Marshall Space Flight Center has developed

NASA's technical standards (MSFC-STD-3176 and MSFC-SPEC-3717) for the design and quality assurance processes associated with additive manufacturing.

The NRC will continue to work with the Department of Energy, Office of Nuclear Energy (DOE-NE) through the Memorandum of Understanding (MOU) between NRC and DOE on Cooperative Nuclear Safety Research and include a new addendum for advanced manufacturing research activities through DOE-NE's Advanced Methods for Manufacturing (AMM) program. The NRC and EPRI will share research and knowledge through quarterly coordination meetings, possible technical staff exchanges, and participating in relevant conferences and meetings.

The NRC also intends to initiate interaction with the Department of Defense (DOD) on their implementation of components made using AM. The Navy currently has components in-service and is developing specifications to govern the fabrication and use of AM components in multiple applications. Because of similar materials, the NRC intends to leverage publicly available research performed by the Navy to better develop regulations for the commercial nuclear industry.

NIST has a well-established research program evaluating a variety of AM processes for polymers and metals. They are documenting the essential variables of each process and resulting properties, such as strength and fatigue. Currently, they have little or no data on the performance of AM materials in NPP environments (i.e., aqueous corrosion, radiation). The NRC intends to engage with NIST to share research and aid in the development of regulations for AM.

C.3 Industry Groups

Electric Power Research Institute (EPRI)

The NRC will continue to work with the Electric Power Research Institute, Inc. (EPRI) to collaborate on research activities on AMTs. Currently, EPRI is performing research on multiple AMTs, at least some of which is a part of the DOE AMM program. The NRC and EPRI will share research and knowledge through:

- Creation of an addendum to the EPRI-NRC Memorandum of Understanding for Cooperative Nuclear Safety Research pertaining to AM,
- Participate in quarterly conference calls to provide updates on research progress, and,
- Participate in conferences and meetings pertaining to AMTs.

NSSS Vendors and OEMs

NRC has been in contact with major nuclear vendors (GE Hitachi, Westinghouse) to discuss their development activities related to AMT, and their plans to deploy AMT in nuclear power plants. GE Hitachi has made several prototype SSCs, with their primary focus on a fuel assembly inlet filter. AM enabled design modifications optimize the pressure drop and particle size screening of the filter which could not previously be obtained by conventional machining. Although GE is making AM parts for production gas turbines, they made a corporate decision to cut back and delay their AM program for nuclear parts.

Westinghouse has researched the properties of stainless steel in AM builds, including in exposure to elevated temperatures and radiation. While they also have produced several prototype parts, their focus has been on a thimble tube plugging device as a demonstration component.

C.4 International Organizations

The NRC is cognizant of extensive work being done on reactor-relevant AMTs internationally, although at this point not actively engaged. Scientists from the Chinese Institute of Nuclear Energy Safety Technology have applied additive manufacturing to manufacturing complex components in nuclear energy systems. Siemens has a project to create 3D-printed pump impeller replacement parts to extend the life of a Slovenian NPP. Russia's nuclear energy corporation, Rosatom, is in advanced stages of implementing an industrial-scale 3D printer.