## **AMP 104 REACTOR HEAD CLOSURE STUD BOLTING (VERSION 2017)**

### Programme Description

This AMP includes in-service inspections (ISI) and preventive measures to detect and manage cracking and loss of material of the closure bolting components of the reactor vessel head. The reactor head closure stud bolting includes the studs, washers, bushings, nuts, and flange threads.

### Evaluation and Technical Basis

1. ***Scope of the ageing management programme based on understanding ageing:***

This AMP manages the ageing effects of cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), strain-induced corrosion cracking (SICC), corrosion fatigue, or environmentally assisted fatigue (EAF) for reactor vessel closure stud bolting for both BWRs and PWRs. This AMP also manages the ageing effects of loss of material due to wear or corrosion for reactor vessel closure stud bolting for BWRs and PWRs. In the United States, Regulatory Guide (RG) 1.65 [1], Revision 1, provides guidance on selecting reactor vessel closure stud bolting materials and properties, conducting a preservice inspection, and conducting ISI.

1. ***Preventive actions to minimize and control ageing degradation:***

Preventive measures may include:

* Avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement;
* Using manganese phosphate or other acceptable surface treatments;
* Using stable lubricants. Of particular note, use of molybdenum disulfide (MoS2) as a lubricant has been shown to be a potential contributor to SCC and is not used [1]; and
* Using bolting material for closure studs that has an actual measured yield strength less than 1,034 MPa (NUREG-1339 [2]). Bolting material with a tensile strength below 1,172 MPa is also relatively resistant to SCC [3].

Implementation of these measures can reduce the potential for SCC, IGSCC, or SICC to occur.

1. ***Detection of ageing effects:***

The extent and schedule of the inspection and test techniques prescribed by the programme are designed to maintain structural integrity and ensure that ageing effects are detected and managed through corrective actions to adequately maintain the intended function of the component. Inspections can reveal cracking, loss of material due to corrosion or wear, and leakage of coolant.

This AMP uses visual, surface, and/or volumetric examinations, in accordance with the pertinent governing requirements (e.g., ASME Code, Section XI [4]) and guidance documents for the plant, to detect the applicable ageing effects. These exams are described in IAEA-TECDOC-1470 [5] and IAEA-TECDOC-1556 [6]. Surface examinations use a magnetic particle or liquid penetrant examination method to indicate the presence of surface discontinuities and flaws. Volumetric examinations use a radiographic or ultrasonic examination method to indicate the presence of discontinuities or flaws throughout the volume of material. Visual VT-2 examinations detect evidence of leakage from pressure retaining components, as required during the system pressure test.

Consistent with the governing national Codes and standards, pressure retaining components are subject to visual VT-2 examinations during the system leakage test (e.g. [4]). IAEA-TECDOC-1470 [5] and IAEA-TECDOC-1556 [6], Section 5, or pertinent governing requirements or guidance documents for the plant address the guidelines for the inspections, monitoring, and maintenance.

1. ***Monitoring and trending of ageing effects:***

This programme has no specific monitoring and trending activities.

1. ***Mitigating ageing effects:***

Actions to mitigate ageing effects are the same as the preventive actions described in attribute 2. OECD/NEA/CSNI/R (2010)15 [7] provides additional guidance.

1. ***Acceptance criteria:***

Acceptance criteria are provided by pertinent governing requirements or guidance documents for the plant. Examples of acceptance criteria are as follows:

* The Inservice Inspection (ISI) programme detects and sizes cracks, detects loss of material, and detects coolant leakage by the examination and inspection as specified in the applicable national Codes [4, 8] and regulations [9].
* Any indication or relevant condition of degradation in closure stud bolting is evaluated in accordance with the acceptance standards defined in the applicable national Codes and regulations.

In addition, IAEA-TECDOC-1470 [5] provides guidelines for the evaluation of detected degradation.

1. ***Corrective actions:***

Repair and replacement activities are performed in accordance with the pertinent governing Codes and requirements or guidance documents for the plant. Regulatory Guide (RG) 1.65 [1], Revision 1, provides guidance on selecting reactor vessel closure stud bolting materials, material properties, and conducting a preservice inspection.

1. ***Operating experience feedback and feedback of research and development results:***

This AMP addresses the industry-wide generic experience. Relevant plant-specific operating experience is considered in the development of the plant AMP to ensure the AMP is adequate for the plant. The plant implements a feedback process to periodically evaluate plant and industry-wide operating experience and research and development (R&D) results, and, as necessary, either modifies the plant AMP or takes additional actions (e.g. develop a new plant-specific AMP) to ensure the continued effectiveness of the ageing management.

SCC has occurred in BWR pressure vessel head studs [10]. Degradation of threaded closure bolting and fasteners for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue (NRC Inspection and Enforcement Bulletin 82-02 [11], NRC Generic Letter 91-17 [12]). Additional information on operating experience can be found in OECD/NEA/CSNI/R (2010)15 [7].

At the time when this AMP was produced, no relevant R&D was identified.

1. ***Quality management:***

Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the different national regulatory requirements (e.g., 10 CFR 50, Appendix B [13]).

### References

1. UNITED STATES NUCLEAR REGULATORY COMMISSION, Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, Revision 1, USNRC, April 2010.
2. UNITED STATES NUCLEAR REGULATORY COMMISSION, Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants, NUREG-1339, USNRC, June 1990.
3. Gross, J.H. “The Effective Utilization of Yield Strength,” Transactions of the American Society of Mechanical Engineers, Paper No. 7, Pressure Vessel and Piping Conference (PVP)-11, Vancouver, British Columbia, Canada, July 23-27, 2006.
4. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Rules for Inservice Inspection of Nuclear Power Plant Components, The ASME Boiler and Pressure Vessel Code, ASME Section XI, as approved in 10 CFR 50.55a, ASME, New York, NY.
5. INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and management of ageing of major　nuclear power plant components important to safety: BWR pressure vessels, IAEA-TECDOC-1470, IAEA, October 2005.
6. INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR pressure vessels, IAEA-TECDOC-1556, IAEA, 2007.
7. NUCLEAR ENERGY AGENCY, Technical Basis for Commendable Practices on Ageing Management-SCC and Cable Ageing Project (SCAP) Final Report, OECD/NEA/CSNI/ R(2010)15, NEA, Paris, April 2011.
8. JAPAN SOCIETY OF MECHANICAL ENGINEERS, IA, IB Code for Nuclear Power Generation Facilities - Rule on Fitness-for-Service for Nuclear Power Plants - JSME S NA1 -2008, JSME.
9. UNITED STATES NUCLEAR REGULATORY COMMISSION, 10 CFR 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, USNRC, Latest Edition.
10. Stoller, S. M., Reactor Head Closure Stud Cracking, Material Toughness Outside FSAR - SCC in Thread Roots, Nuclear Power Experience, BWR-2, Ill, 58, p. 30, 1991.
11. UNITED STATES NUCLEAR REGULATORY COMMISSION, Bulletin 82-02: Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants, USNRC, May 1986.
12. UNITED STATES NUCLEAR REGULATORY COMMISSION, Generic Letter 91-17, Bolting Degradation or Failure in Nuclear Power Plants, USNRC, October, 1991.
13. UNITED STATES NUCLEAR REGULATORY COMMISSION, 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, USNRC, Latest Edition.