UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 25, 1998

NRC INFORMATION NOTICE 98-11: CRACKING OF REACTOR VESSEL INTERNAL BAFFLE FORMER BOLTS IN FOREIGN PLANTS

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs) except those who have permanently ceased operation and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the cracking of reactor vessel internal baffle former bolts (see Figures 1 and 2) found at several foreign PWRs and to inform addressees of actions taken and planned by domestic PWR owners groups in response to this experience. It is expected that the recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Reactor vessel internals are structures located within the reactor vessel that support and orient the reactor fuel assemblies and direct coolant flow through the core. The core baffle is part of the internals structure, which consists of vertical plates that surround the outer faces of the peripheral fuel assemblies. The baffle directs coolant flow through the core. The vertical plates are bolted to the edges of horizontal former plates that are bolted to the inside surface of the core barrel. There are typically eight levels of former plates located at various elevations within the core barrel. The bolts that secure the baffle plates to the former plates are referred to as "baffle former bolts."

European plants identified the cracking of baffle former bolts as early as 1988 and this problem continues to occur. Although this cracking is not fully understood, testing of cracked bolts suggests an age-related intergranular stress-corrosion cracking process influenced by bolt material, fluence, stress, and temperature. The reported cracking occurred in 316 cold-worked stainless steel bolts. Most of the cracking reported has been in four French 900-MWe (megawatt electric) PWRs.

An investigation of the cracking was discussed in a paper contained in the Proceedings of the International Symposium Fontevraud III, dated September 12-16, 1994, held at the Royal Abbey of Fontevraud, France. The symposium paper reports that the cracking of baffle former bolts seems to be limited to the first six PWRs operated by Electricité de France (EDF), which are all of the same design and are identified as the "CPO" series. Further, the paper reports that bolt cracking has not been seen in the other French 900-MWe plants (the "CPY" series), or in the 1300-MWe plants. The paper notes that there are differences between the two series with regard to bolt design, bolt material, operating conditions, and reactor coolant flow paths. Some plants in both groups have been in operation for approximately the same number of hours. The plants which reported the greatest number of cracked bolts are Fessenheim Unit 2 and Bugey Unit 2, both of which are CPO series plants. The number of cracked bolts identified at these plants are 29 and 54, respectively. All of the baffle former bolts (960) in each plant were tested ultrasonically.

Discussion

At the foreign plants, ultrasonic testing was performed to assess the integrity of the baffle former bolts. Five bolts were removed from the Bugey Unit 2 baffle assembly for a detailed investigation of the degradation process. One of the bolts was found to be broken, three were found to be cracked, and one was found to be sound. The conclusions reached in the symposium paper are that (1) baffle bolt cracking has been limited to plants of the same design (CPO series), (2) bolt cracking has occurred predominantly in zones of fluence and maximum temperature, (3) in the zones of maximum fluence, bolt cracking is found predominantly in the bolts under the highest mechanical stress, and (4) bolt cracking has occurred in some plants although not in other plants of the same design, a phenomenon that may be a consequence of varying bolt metallurgical properties and plant operating conditions.

The NRC is not aware of cracking of baffle former bolts in domestic PWRs. Domestic reactor baffle former bolts are subject to the visual inservice inspection requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. However, the baffle bolt cracking reported in foreign PWRs has occurred at the juncture of the bolt head and the shank, which is not accessible for visual inspection.

Domestic PWR owners groups have met with the NRC staff to report on their current and planned activities regarding the potential for baffle bolt cracking in domestic PWRs. The details of those meetings are discussed below.

The Westinghouse Owners Group (WOG) provided an assessment of the cracking of the baffle former bolts identified in foreign PWRs, including the potential impact of cracking on domestic Westinghouse plants, and provided information on its current and planned activities. The WOG stated that because of the large number of baffle former bolts in the baffle assembly, the failure of a few bolts should not have a significant safety impact. The WOG activities include (1) development of analytical methods and acceptance criteria for bolt analysis, (2) performance of risk-informed evaluations, (3) performance of analysis for three plant groupings (2-loop, 3-loop, and 4-loop) of what constitutes acceptable bolting, (4) continued participation in domestic and foreign related activities, (5) preparation of bid specifications for bolt inspection equipment, and (6) preparations for bolt inspection and replacement. The WOG identified lead plant candidates for the 2-loop and 3-loop groups and a proposed inspection schedule for each group. The WOG indicated that the bolt material used in the 2-loop group is 347 stainless steel and the bolt material used in the 3-loop group is 316 cold-worked stainless steel.

The Babcock and Wilcox (B&W) Owners Group (B&WOG) provided information on its current and planned activities to address the potential for cracking of baffle former bolts in domestic B&W plants, including a presentation of its "Plant Licensing Reactor Vessel Internals Aging Management Program." The B&WOG provided a preliminary determination that bolt cracking is not considered a significant safety issue for B&W plants. This determination is based upon knowledge of the baffle and bolting designs involved and is supported by conservative analyses that assume both normal-operating and maximum accident-loading conditions. The B&WOG activities include (1) collection and evaluation of available inspection and material data, (2) development and qualification of replacement bolt materials, and (3) preparation of a possible baffle bolt inspection on a lead plant during the next 10-year inservice inspection interval. The B&WOG indicated that the bolt material used in B&W plants is 304 stainless steel.

The Combustion Engineering (CE) Owners Group (CEOG) provided an assessment of the cracking of the baffle former bolts reported in foreign PWRs, including the potential impact of the cracking on domestic CE plants. The CEOG believes that the most likely mechanism for the cracking of cold-worked 316 stainless steel baffle former bolts in foreign plants is irradiation-assisted stress-corrosion cracking (IASCC). The CEOG indicated that only two of its plants use bolts to attach the core shroud panels (i.e., the baffle plates) to the former plates. The CEOG believes that these two plants are less susceptible to IASCC because of several design differences: (1) the material used in these bolts is annealed 316 stainless steel, which is not cold worked; (2) the bolt stress from preload, as a percentage of yield strength, is much less than the EDF plants; (3) the differential pressure across the core shroud panels does not result in tensile loads on the panel (i.e., the baffle) bolts during normal operation; and (4) the core shroud panel design allows for some flexing of the former plate relative to the core barrel, thus effectively reducing the load on the panel bolts.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

for Jack W. Roe, Acting Director
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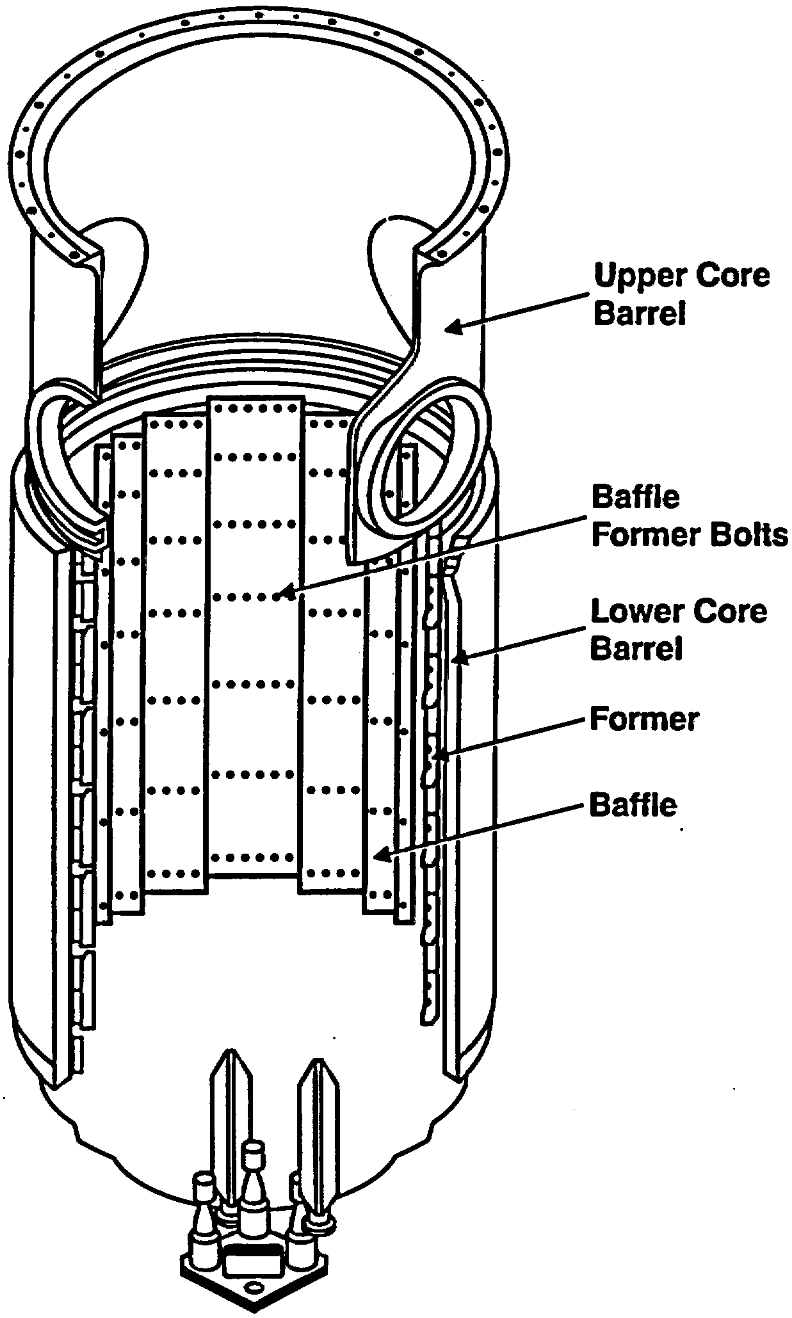
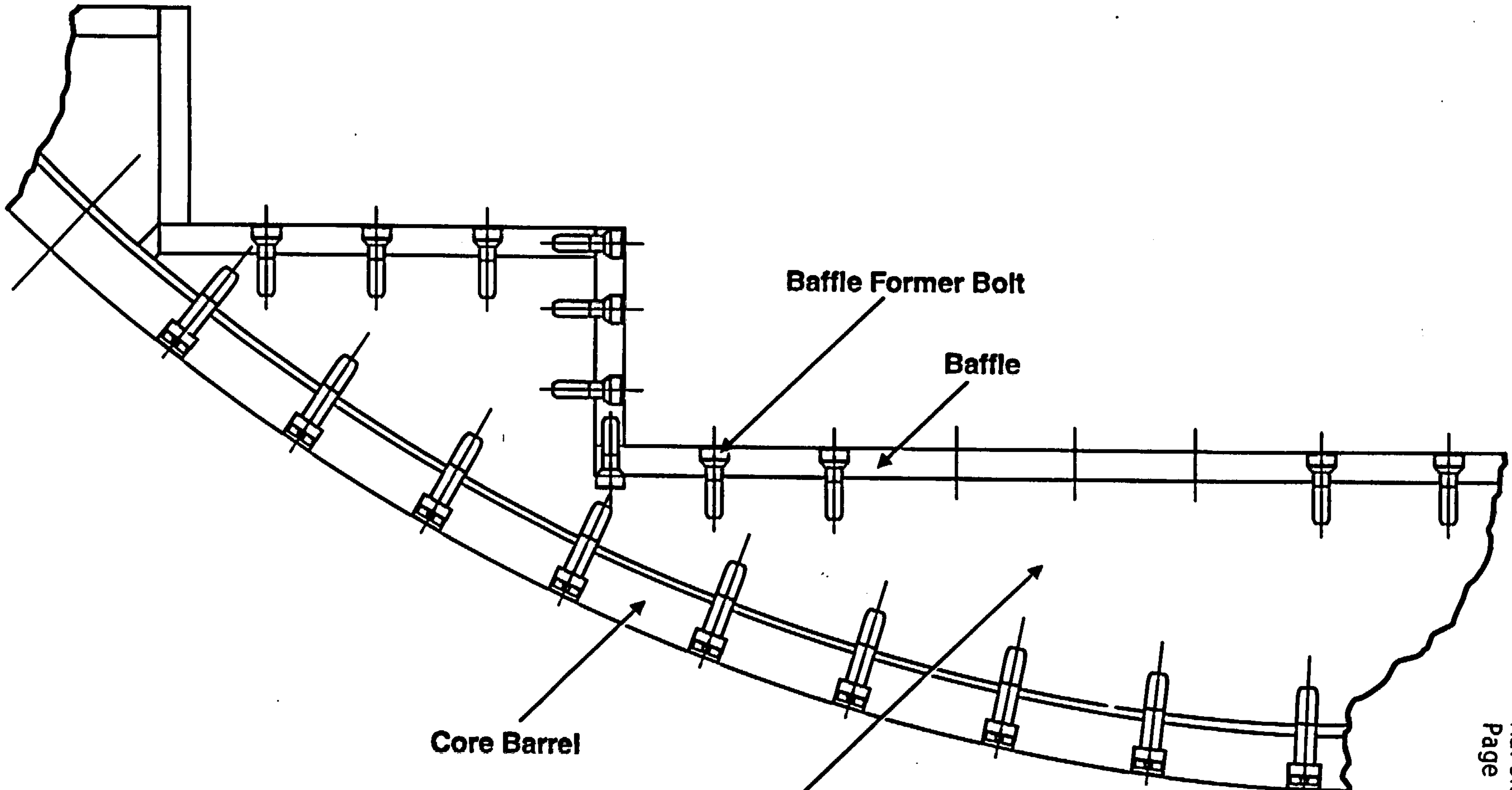
Attachments:
1. Typical PWR Reactor Vessel Internals
2. Typical PWR Core Barrel, Former, and Baffle Arrangement
3. List of Recently Issued Information Notices



Typical PWR Reactor Vessel Internals



Typical PWR Core Barrel, Former, and Baffle Arrangement

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
95-52, Supp.1	Fire Endurance Test Results for Electrical Raceway Fire Barrier Systems Constructed From 3M Company Interam Fire Barrier Materials	3/17/98	All holders of operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the reactor vessel.
98-10	Probable Misadministrations Occurring During Intravascular Brachytherapy With The Novoste Beta-Cath System	3/9/98	All Medical Licensees
98-09	Collapse Of An Isocam II Dual-Headed Nuclear Medicine Gamma Camera	3/5/98	All Medical Licensees
98-08	Information Likely To Be Requested If An Emergency Is Declared	3/2/98	All Parts 30, 40, 70, 72, and 76 licensees and certificate holders required to have a Nuclear Regulatory Commission-approved Emergency Plan.
98-07	Offsite Power Reliability Challenges from Industry Deregulation	2/27/98	All holders of operating licensees for nuclear power reactors
98-06	Unauthorized Use of License to Obtain Radioactive Materials, And Its Implications Under The Expanded Title 18 of the U.S. Code	2/19/98	All NRC licensees authorized to possess licensed material
97-45, Supp. 1	Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails	2/17/98	All holders of operating licenses for nuclear power reactors except those licensees who have permanently ceased operations and have certified that the fuel has been permanently removed from the reactor vessel

OL = Operating License
CP = Construction Permit