MAR 0 8 1978

Docket No.: 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. Gerlad K. Rhode
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

SUBJECT: NRC STAFF POSITION ON THE USE OF AUSTENITIC STAINLESS STEEL IN BOILING WATER REACTOR FACILITIES-NINE MILE POINT NUCLEAR STATION, UNIT 2

During the past several years, the NRC and its predecessor agency, the AEC, have conducted an extensive investigation to evaluate the cracking of austenitic stainless steel piping. This effort was initiated following the detection in late 1974 and early 1975 of a series of cracks in the piping of boiling water reactor facilities. As a result of this investigation, we have concluded that the types of austenitic stainless steel currently used in boiling water reactor piping are susceptible to stress corrosion cracking.

The staff believes the probability is extremely low that such stress corrosion cracks will propagate far enough to create a significant safety hazard to the public. However, we have also concluded that steps should be taken to eliminate this condition. To this end, we have developed a position to set forth acceptable methods to reduce the susceptibility of boiling water reactor piping to stress corrosion cracking. This position is contained in NUREG-0313, dated July 1977, a copy of which is enclosed. We have also incorporated the position contained in NUREG-0313 as Branch Technical Position MTEB 5-7 and issued it as a revision to the Standard Review Plan.

You should note that the implementation schedule set forth in the position provides for varying degrees of conformance, depending upon the status of the application. We require that you provide a schedule for your response to this position within 14 days of receipt of this letter. Your response should address each of the subsections in Section II and III of the position. Forty (40) copies of your response are needed for use by the staff.

OFFICE➤						
SURNAME➤						
DATE➤						

If you require any clarification of this request, please contact the staff's assigned Licensing Project Manager.

This request for generic information was approved by GAO under a blanket clearance No. R0071. This clearance expires September 30, 1978.

Sincerely,

Original signed by C. Stahle for/

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosure:
NUREG-0313, dated
July 1977

cc w/enclosure:
See page 3

OFFICE➤	DPM:LWR #4	DPM:LWR #4				
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NUREG-0313

TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING

Manuscript Completed: July 1977
Date Published: July 1977

Division of Operating Reactors
Division of Systems Safety
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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I. INTRODUCTION

Small, hairline cracks in austenitic stainless steel piping in boiling water reactor (BWR) facilities were observed as early as 1965. In each case, it was believed that the situation had been corrected or substantially reduced by better control of welding, contaminants and/or design modifications. In September, 1974, when the first of a series of cracks in the piping of the more modern BWRs was found at Dresden Unit No. 2., the then Atomic Energy Commission (AEC) initiated an intensive investigation to evaluate the cause, extent, and safety implications of the observed cracking. In January 1975, a special Pipe Cracking Study Group was formed to coordinate and accelerate the staff's continuing investigations of the occurrences of pipe cracking. This group included representatives of the Nuclear Regulatory Commission (NRC) and their consultants. In October, 1975, the Study Group issued a report, NUREG-75/067 "Technical Report, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants." During the same general time span, the General Electric Company (GE) conducted an independent evaluation of the cracking occurrences and submitted their findings and recommendations to the NRC. This paper sets forth the NRC technical position based on the information available at this time.

Plant operating history indicates that Type 304 and 316 austenitic stainless steel piping in the reactor coolant pressure boundary of boiling water reactors are susceptible to stress corrosion cracking.

Studies have shown that such cracking is caused by a combination of the presence of significant amounts of oxygen in the coolant, high stresses, and some sensitization of metal adjacent to welds. Such cracks have occurred in the heat affected zones adjacent to welds but are not expected to occur outside these areas, provided that the pipe material is properly annealed.

Pipe runs containing stagnant or low velocity fluids have been observed to be more susceptible to stress corrosion cracking than pipes containing a continuously flowing fluid during plant operation. Historically, these cracks have been identified either by volumetric examination, by leak detection systems, or by visual inspection. Because of the inherent high material toughness of austenitic stainless steel piping, stress corrosion cracking is unlikely to cause a rapidly propagating failure resulting in a loss-of-coolant accident.

Although the probability is extremely low that these stress corrosion cracks will propagate far enough to create a significant safety hazard to the public, the presence of such cracks is undesirable. Steps should therefore be taken to minimize stress corrosion cracking in BWR piping systems to eliminate this condition and to improve overall plant reliability.

It is the purpose of this position to set forth acceptable methods to reduce the stress corrosion cracking susceptibility of BWR piping and thereby also provide an increased level of reactor coolant pressure boundary integrity. Recognizing that the most straightforward and

desirable approach or methods may not be practicable, or even possible, for all plants, the bases for varying degrees of conformance to our guidelines are provided. Augmented inservice inspection and leak detection requirements are established for plants that have not fully implemented the provisions contained in Part II of this document.

II. SUMMARY OF ACCEPTABLE METHODS TO MINIMIZE CRACK SUSCEPTIBILITY

The material selection and processing guidelines listed below identify alternative acceptable methods to minimize susceptibility to stress corrosion in BWR pressure boundary piping. It is expected that adoption of these practices will result in a high degree of protection against stress corrosion cracking.

1. Corrosion Resistant Materials

   All pipe and fitting material including weld metal should be of a type and grade that has been shown to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition. Unstabilized wrought austenitic stainless steel with $\geq$0.035% carbon does not meet this requirement unless all such piping including welds is in the solution annealed condition. The acceptability of alternative materials, processes, or other methods to provide an adequate degree of corrosion resistance will be made on a case-by-case basis.

2. Corrosion Resistant "Safe Ends"

   All unstabilized wrought austenitic stainless steel piping with carbon contents $\geq$0.035% should be in the solution annealed condition.

If welds joining these materials are not solution annealed, they should be made between case (or weld overlaid) austenitic stainless steel surfaces (5% minimum ferrite) or other materials having high resistance to oxygen-assisted stress corrosion. The joint design must be such that any unstabilized wrought austenitic stainless steel containing $\geq$0.035% carbon, which may become sensitized as a result of the welding process, is not exposed to the reactor coolant.

3. Other proposed methods to provide protection against stress corrosion cracking will be reviewed on a case by case basis.

Regulatory Guide 1.44 "Control of the Use of Sensitized Stainless Steel", dated May, 1973 will be revised to provide additional guidance on acceptable practices.

III. INSERVICE INSPECTION AND LEAK DETECTION REQUIREMENTS FOR BWRs WITH VARYING CONFORMANCE TO MATERIAL SELECTION AND PROCESSING GUIDELINES

1. For plants where all ASME Code Class I reactor coolant pressure boundary piping subject to inservice inspections under Section XI meets the guidelines stated in Part II, no augmented inservice inspection or leak detection requirements are necessary.

2. Piping in all other plants is subject to additional inservice inspection and leak detection requirements, as described below. The degree of inspection of such piping depends on whether the specific piping runs are conforming or non-conforming, and on whether the specific piping runs are classified as "Service

Sensitive". "Service Sensitive" lines are defined as those that have experienced cracking in service, or that are considered to be particularly susceptible to cracking because of high stress, or because they contain relatively stagnant, intermittent, or low flow coolant.

Examples of piping runs considered to be service sensitive include, (but are not limited to): core spray lines, recirculating by-pass lines (or "stub tubes" on plants that have removed the by-pass lines) CRD hydraulic return lines, isolation condenser lines, and shut down heat exchanger lines.

A. For non-conforming lines that are not service sensitive:

(1) Inservice inspection of the non-conforming lines should be conducted in accordance with the schedule specified in ASME Code, Section XI - Subsection IWB, as required by the applicable examination Categories B-F and B-J, with the exception that the required examination should be completed in no more than 80 months (two thirds of the time perscribed in the schedule in the ASME Boiler and Pressure Vessel Code Section XI). If examinations conducted during the first 80 month period reveal no incidence of stress corrosion cracking, the examination schedule thereafter can revert to the schedule perscribed in Section XI of the ASME Boiler and Pressure Vessel Code.

The piping areas subject to examination, the method of examination, the allowable indication standards and examination procedures should comply with the requirements of the Edition and Addenda of the ASME Code, Section XI identified as applicable by 10 CFR Part 50, Section 50.55a, Paragraph (g), "Codes and Standards."

(2) The reactor coolant leakage detection system should be operated under the following Technical Specification requirements in order to enhance the discovery of unidentified leakage that may include through-wall cracks developed in austenitic stainless steel piping:

a. The source of reactor coolant leakage should be identifiable to the extent practical, using leakage detection and collection systems that meet the position described in Section C, Regulatory Position of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," or an acceptable equivalent system.

b. Plant shutdown should be initiated for inspection and corrective action when the leakage system indicates, within a period of four hours or less, an increase in the rate of unidentified leakage in excess of two gallons per minute, or when the total unidentified leakage attains a rate of five gallons per minute, whichever occurs first.

c. Unidentified leakage should include all leakage other than:

1. Leakage into closed systems, such as pump seal or valve packing leakage that is captured, metered, and conducted to a sump or collecting tank,

2. Leakage into the containment atmosphere from sources that are specifically located and known either not to interfere with the operation of the unidentified leakage detection system, nor not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

B. For non-conforming lines that are service sensitive:

(1) The leakage detection requirements described in III.A above, should be implemented.

(2) The welds and adjoining areas of bypass piping of the discharge valves in the main recirculation loops, and of the austenitic stainless steel reactor core spray piping up to and including the second isolation valve should be examined at each reactor refueling outage or at other scheduled or unscheduled plant shutdowns. Successive examinations need not be closer than six months, if shutdowns occur more frequently than six months. This requirement applies to all bypass lines whether the 4-inch valve is kept open or closed during operation.

In the event these examinations find the piping free of unacceptable indications for three successive inspections, the examination may be extended to each 36 month period (plus or minus by as much as 12 months) coinciding with a refueling outage. In these cases, the successive examination may be limited to one bypass pipe run, and one reactor core spray piping run.

(3) The welds and adjoining areas of other service sensitive piping should be examined on a sampling basis. For example, if a system consists of several branch runs with essentially symmetric piping configurations that perform similar system functions, an acceptable inspection program should include at least one, but not less than 25%, of the similar branch runs. The frequency of such examinations should be as described in 2 above. If unacceptable flaw indications are detected in any branch run, the remaining branch runs among the group should be examined.

In the event the examinations reveal no unacceptable indications within three successive inspections, the examination schedule may revert to the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components" with the exception that the required examination should be completed during each 80 month period (two-thirds the time perscribed in the schedule in the ASME Code Section XI).

(4) The method of examination, the allowable indication standards and examination procedures should comply with the requirements of the Edition and Addenda of the ASME Code, Section XI identified as applicable by 10 CFR Part 50, Section 50.55a, Paragraph (g), "Codes and Standards."

IV. IMPLEMENTATION OF MATERIAL SELECTION AND PROCESSING GUIDELINES

1. For plants that apply for a construction permit after the issue date of this document, all ASME Code Class I reactor coolant pressure boundary lines should conform to the guidelines stated in Part II.*

2. For plants under review, but for which a construction permit has not yet been issued, all service sensitive lines should conform to the guidelines stated in Part II. Other ASME Code Class I reactor coolant pressure boundary lines should conform to Part II to the extent practicable.

3. For plants that have been issued a construction permit, ASME Code Class I reactor coolant pressure boundary lines should conform to the guidelines stated in Part II to the extent practicable.

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*After revision, Regulatory Guide 1.44 may be used as guidance for acceptable materials, process, or other methods.

4. For plants that have been issued an operating license, service sensitive lines should be modified to conform to the guidelines stated in Part II, to the extent practicable. Lines in which cracking is experienced should be replaced with piping that conforms to the guidelines stated in Part II.

V. GENERAL RECOMMENDATIONS

The measures outlines in Part II of this document provide for positive actions that are consistent with the current technology. The implementation of these actions should markedly reduce the susceptibility to stress corrosion cracking in BWRs. It is recognized that additional techniques are available to limit the corrosion potential of BWR coolant pressure boundary materials and improve the overall system integrity. These include plant design and operational considerations to reduce system exposure to potentially aggressive environment, improve material fabrication and welding techniques and provisions for volumetric inspection capability in the design of weld joints. Specifically, consideration should be given to:

1. Minimizing the total extent of the coolant pressure boundary with special emphasis on stagnant or low flow lines.

2. Reducing the oxygen content of the primary coolant.

MAR 0 8 1978

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