NUCLEAR REGULATORY COMMISSION

August 5, 1981

MEMORANDUM FOR: R. H. Vollmer, Director
Division of Engineering

FROM: J. Halapatz
Materials Engineering Branch

SUBJECT: DIFFERING PROFESSIONAL OPINION RELATED GENERICALLY TO SENSITIZATION OF BWR STAINLESS STEEL WELDMENTS

REFERENCE: Memorandum, W. S. Hazelton to R. H. Vollmer, dated July 31, 1981, subject, "Differing Professional Opinion Related to Sensitization of BWR Stainless Steel Weldments"

This memorandum is identified as a statement for record in accordance with NRC Appendix 4125. This memorandum addresses the reference memorandum:

Mr. Hazelton:

"Although most of the components he mentions are generally included in reactor internals, he also mentions installed large diameter pipe and concrete embedded flued heads. I don't know which systems he is referring to, however. My reviewers and I have some questions and concerns regarding how this 'issue' will be resolved."

Halapatz responds:

The large diameter pipe and concrete embedded flued heads mentioned are those nonconforming materials for which licensees will claim undue hardship related to replacement thereof within the context of NUREG-0313, Rev. 1. The Pennsylvania Power and Light Program for mitigation of IGSCC at Susquehanna, for example, see attachment 1, claims undue hardship related to such components. The large pipe is in the recirc and RHR systems. The flued heads are in the reactor water cleanup and core spray systems. Given that these materials are nonconforming, but will not be replaced, and with an undetermined accessibility for inservice inspection, it follows that metallurgical characterization of these materials with respect to propensity to IGSCC is in order.

Mr. Hazelton:

"1. First, what is the issue? We've been going around in circles on this one, but if it is to be resolved by a trip to San Jose, it appears to me that we'd better make sure that we know what it is that needs to be resolved.

a. If the question is whether some welds are, or could be sensitized, we can stipulate right now to that effect.

b. If the question is whether all GE supplied equipment met or does meet Regulatory 1.44, we know that the answer is negative.

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c. If the question is whether components not made to 1.44, and/or are sensitized, are likely to fail in service, looking over welding procedures will not resolve the issue."

Halapatz responds:

1. The issue is that we really haven't metallurgically characterized the materials of interest with respect to their propensity to IGSCC. Halapatz assures Mr. Hazelton that Halapatz hasn't been going around in circles on this one. Halapatz has been to San Jose before; he takes no great delight in traveling on business to San Jose or elsewhere. should that be bothering Mr. Hazelton, but he thinks we simply must take a good hard look to determine what materials are in place in NTOLS and what their propensity is to IGSCC based on guidelines and criteria the NRC applies in the licensing process, i.e., we should anticipate rather than commiserate failures. We should then implement a well-defined surveillance program for these components identified as IGSCC prone.

   a. It's the severity of sensitization That's relevant. It's [illegible] accepted that higher heat input develops more severe sensitization and a greater propensity to IGSCC. This position is endorsed by R.G. 1.44, attachment 2, via the following:

   "All welding processes will result in some carbide precipitation in the weld metal and in base metal heat-affected zone of stainless steel welds, but significant sensitization does not normally result when typical welding procedures and material chemistry are used and when no further heating of material occurs. However, there is evidence that atypical welding methods using very high heat input could result in stress corrosion cracking in the heat-affected zone of the weld. To avoid this, the welding procedures and material chemistry (if necessary) should be controlled to prevent undue sensitization of the heat-affected zones of the weldments. Controls to prevent sensitization of the material during welding may include: (1) avoiding welding practices that result in the generation of high heat, (2) maintaining low heat input by controlling current, voltage, and travel speed, (3) limiting interpass temperature, (4) using stringer bead techniques and avoiding excessive weaving, and (5) limiting the carbon level of the material where section thickness makes the material more prone to sensitization.

   "In addition, welding procedures[2] should be qualified by passing a suitable intergranular corrosion test in all cases where the procedure is used for welding stainless steel having a carbon level greater than 0.03 percent. The qualification test should be performed using base material having the maximum carbon content anticipated and the minimum and maximum thicknesses anticipated.

"As a minimum, the variables that should be controlled in the qualification test are heat input, interpass temperature, and welding techniques for specific section thicknesses."

"6. Welding practices and, if necessary, material composition should be controlled to avoid excessive sensitization of base metal heat-affected zones of weldments. An intergranular corrosion test, such as specified in subparagraph C.3. above, should be performed for each welding procedure to be used for welding material having a carbon content of greater than 0.03 percent."

This position also is endorsed by the current GE position on compliance with R.G. 1.44 as developed within GESSAR Docket No. STN-50-447. As shown in attachment 3, Shoreham, for example, committed for NSSS to the GE position in GESSAR, Docket STN-50-447, which limited welding heat input to 110,000 joules/in. GE, however, later reconsidered, in 1980, its position as shown by the GESSAR II Docket 50-447 and limited welding heat input to 50,000 joules/in., in essential compliance with R.G. 1.44.

The differing professional opinion expressed is consistent with R.G. 1.44 and the current GE position.

To the extent that it can be established, however, welding was performed on Shoreham components to the 110,000 joule/in limit. On this basis, suspect welds having a greater propensity to IGSCC should exist. NUREG-0313, Rev. 1, invoked by the NRC, however, addresses only piping, but not reactor internals or structurals nor nonconforming materials as cited for Susquehanna.

b. Mr. Hazelton's statement that GE equipment is in place which does not meet R.G. 1.44 is a major contradiction and serious challenge to the credibility of MTEB SERs which address the matter. The Shoreham SER, Docket 50-322, dated April 1981, attachment 4, for example, acknowledges that R.G. 1.44 has been met, despite the farcical use of the expression "satisfy the intent of R.G. 1.44."

   The LaSalle FSAR commitment to the 110,000 joules/in. limit indicates that the issue is generic (attachment 5).

c. Halapatz, R.G. 1.44, and currently GE, share a common position; to wit, that weld heat input should be controlled to lessen the severity of sensitization and hence propensity to IGSCC. "Looking over welding procedures" will contribute much information for the metallurgical characterization of the materials of interest and their propensity to IGSCC. The review would simply assess materials within R.G. 1.44 guidelines.

Mr. Hazelton:

"My reviewers and I believe that the lack of problems with IGSCC failures of BWR internals, together with the results of laboratory tests showing the necessity for maintaining stresses at or above yield to cause IGSCC in BWR environment, justify our position that complete conformance to Regulatory Guide 1.44 is not required for us to make a finding that the applicable GDC's are met."

Halapatz responds:

Halapatz is astounded by Mr. Hazelton's statement that there is a lack of problems with IGSCC of BWR internals. Mr. Hazelton and his reviewers are directed to attachment 6, which is a compilation of nuclear power plant experience to which the NRC library suscribes. Mr. Hazelton will note that the compilation describes IGSCC failures of reactor internals components, which have become commonplace. Based on Mr. Hazelton's premise then, given the number of failures which have occurred, stress levels at or above yield would appear to be the rule rather than exception, in reactor internals and structurals. Mr. Hazelton's argument, relating to at or above yield stress levels seems to be specious.

Based on FSAR information, NTOLs such as Shoreham, reactor internals and structural materials will be the same, be welded using the same welding controls and enter service in the same metallurgical condition, experience the same service environment and endure the same stress levels in service, whatever they may be, as the reactor internals and structural materials applied in pre-NTOL plants. Therefore, they can be expected to crack. Halapatz finds it incredible that NTOLs are being licensed with full knowledge that failures are likely to occur. The basic fact underlying his DPO is that these potential failures should be identified and kept under surveillance.

"3. Inspection programs on BWR internals are being carried out. These include spray lines, feedwater spargers, LPCI connectors, jet pump components, shroud head assembly (stand pipes and steam separators), and the steam dryer assembly. This program has discovered cracks in core spray lines and spargers, for example. (Note that not all cracks were in weld HAZs.) In addition to these components that are inspected fairly often, the CRD guide tubes and instrumentation piping are inspected every 10 years, when it can be made accessible. We believe that this inspection program is adequate, and is about the best that can be done."

Halapatz responds:

In all candor, Mr. Hazelton's first statement, more properly should read, "Inspections on some BWR internals are now being carried out." Item 55 of attachment 6 is illustrative of how the problem is being handled. It would appear more factual to state that cracks discovered in core spray lines, spargers, etc., are forcing us to inspect some components. The fact that not all cracks were in weld HAZs leads one to conclude only that severely sensitized base material was applied improperly.

Given this failure history and given that the metallurgical histories and environmental and stress level experience of the materials applied in NTOLs will be the same as those of pre-NTOL materials which have failed, it is most baffling to read, as in the Shoreham SER dated April 1981:

"4.5.2 Reactor Internal and Core Support Materials

The applicant has met the requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50 by assuring that the design, fabrication, and testing of the materials used in the reactor internals and core support structure are of high quality standards and adequate for structural integrity. The controls imposed upon components constructed of austenitic stainless steel satisfy the intent of the recommendations of Regulatory Guide 1.31, 'Control of Ferrite Content in Stainless Steel Weld Metal' and Regulatory Guide 1.44, 'Control of the Use of Sensitized Stainless Steel.'

"The materials used for the construction of components of the reactor internals and core support structures have been identified by specification and found to be in conformance with the requirements of NUREG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

"The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and core support structure will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component integrity.

"The material selection, fabrication practices, examination and testing procedures, and control practice performed in accordance to those recommendations provide reasonable assurance that the materials used for the reactor internals and core support structure will be in a metallurgical condition to preclude service deterioration. Conformance with the requirements of the ASME Code and the intent of the recommendations of the regulatory guides constitutes and acceptable basis for meeting the requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50."

HALAPATZ CONTINUES:

It is noted that Section XI coverage of BWR reactor internals is in the course of preparation. Section XI acceptance standards for BWR core support structures also are in course of development. NRC guidelines relating to PSI and

ISI of reactor internals are, at this time, no better defined than the inspection program identified by Mr. Hazelton. Halapatz declines to judge as adequate an inspection program based on the accessibility of surfaces for inspection. On this basis, sensitive surfaces, which should be inspected, may never be inspected. Attention is called, in this regard, to item 12 of attachment 6, which addresses cracks in the core support guid of Dresden 1. Halapatz further declines to accept such a program, as described by Mr. Hazelton as "about the best that can be done." There is much more that can be done! It follows, more logically, that the identification of IGSCC prone reactor internals, core support structures and components nonconforming to NUREG-0313, Rev. 1, should be inspected on the basis of mandatory accessibility of surfaces as prescribed inspection intervals more restrictive than those prescribed by Section XI.

Mr. Hazelton:

"4. Our reviewers feel strongly that they should be included in any NRC investigation into the validity of their reviews. It is clear that we cannot afford to send a contingent of six or more people to San Jose for this purpose, therefore, we urge that General Electric be given the opportunity to come here to make a formal presentation on the subject. I would be pleased to assist in preparing a proposed agenda, which I believe should cover much more than details of weld procedures."

Halapatz responds:

Halapatz feels even more strongly about the issue and is very strongly dedicated as his duty, in accordance with NRC Manual Chapter 4125-02, to the expression of his differing professional opinion.

At the risk of appearing irreverent, given the apparent sensitivity which has developed, Halapatz suggests that the NRC would be best served were his differing professional opinion be considered outside the jurisdiction of Mr. Hazelton and his reviewers and instead be submitted to an impartial peer review group for review, evaluation and recommendations. Halapatz is aware that the NRC includes in its staff, individuals, not heretofore identified in this matter, for whom he has professional respect for their knowledge of the technical aspects of the issues, in addition to their professional objectivity and integrity.

J. Halapatz
Materials Engineering Branch

cc: W. V. Johnston, DE
S. S. Pawlicki, DE
G. Johnson, DE
W. Hazelton, DE
J. Thomas, NTEU
M. Levy, NTEU

ATTACHMENT 1

PP&L

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NORMAN W. CURTIS
Vice President-Engineering & Construction-Nuclear
770-5381

June 30, 1981

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO NRC GENERIC LETTER 81-03
AND NUREG 0313 "TECHNICAL REPORT ON
MATERIAL SELECTION AND PROCESSING GUIDELINES
FOR BWR COOLANT PRESSURE BOUNDARY PIPING"
ER 100450 FILE 841-9,883
PLA-856 841-13

Attached please find PP&L's response to Generic letter 81-03 and NUREG 0313 .

Very truly yours,

R.J. Sherlin for

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

TEG/mks

Attachment

THE EFFECTIVE ELIMINATION OF IGSCC AT
SUSQUEHANNA STEAM ELECTRIC STATION

Since 1975, when knowledge of intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel in boiling water reactor (BWR) piping was brought to its attention, Pennsylvania Power & Light Company (PP&L) has undertaken an extensive program for the Susquehanna Steam Electric Station to effectively eliminate the possibility of IGSCC-caused downtime, and also to conform to the requirements of NUREG-0313, Rev. 1, and previous publications.

PP&L's program of IGSCC mitigation includes the following methods:

1. Replacement of susceptible materials, where practical, with materials that conform to Section III of NUREG-0313.

A. Recirculation system discharge valve bypass line (4") - replaced with carbon-limited Type 304 SS (low carbon stainless steel that meets the mechanical properties of regular grade Type 304, and has a maximum carbon content of .03%).

B. Core Spray and Head Spray-replaced with carbon limited Type 304.

C. Reactor water clean-up system (RWCU) - replaced with Type 304L SS.

D. Instrument piping and bottom drain - replaced with Type 304L SS.

Any further replacement of nonconforming material would result in what PP&L feels is an undue hardship, because it would involve replacement of already-installed large diameter (20" or larger) piping or flued heads that are imbedded in concrete.

In view of the above reasons, and with consideration for the very extensive effort by PP&L that resulted with the replacement of a considerable amount of nonconforming material, PP&L requests that further material replacement requirements not be imposed (Generic Letter 81-03) because of severe undue hardship.

2. Elimination of lines whose functions are no longer required.

A. Control Rod Drive Hydraulic Return.

3. Use of low carbon corrosion resistant weld build-up (shop method) for field welds.

A. Recirculation system risers.

4. Design improvement to eliminate crevices.

A. Recirculation system inlet safe ends (extensive modification to replace all safe ends and thermal sleeves).

5. Solution heat treatment.

A. Recirculation system risers (shop welds).

In addition to the mitigation methods listed above, PP&L has used several more, some of which are listed under Section V of NUREG-0313, as follows:

6. Dissolved oxygen control

A. During normal plant operation - PP&L has relocated the control rod drive (CRD) pump intake from the condensate storage tank to the condensate makeup/reject line. This results in using CRD water with the lowest oxygen concentration available (essentially water of feedwater quality).

B. During all phases of operation/shutdown except normal operation, oxygen levels are reduced by the use of a mechanical vacuum deaerator. The deaerator is expected to maintain an oxygen content of less than 250 PPB during start-up, hot standby and shutdown.

7. Welding Parameters

A. Block welding was prohibited.

B. Interpass temperature was limited to 350°F maximum.

C. No preheat was used (in excess of a working range of 60°F to 150°F).

8. Ferrite Control

All of the weld metal and all of the type 304/316 castings were specified to have not less than 5% ferrite content. This level is generally recognized as sufficient to provide immunity from the initiation of IGSCC.

Use of induction heating stress improvement (IHSI)

IHSI is a method of eliminating or substantially reducing residual tensile stresses on the inside surface of the piping. These tensile stresses are primarily caused by welding and have been identified as one of the three major causes of IGSCC. PP&L is presently considering the use of IHSI for 108 welds on No. 1 unit that do not conform to Section III of NUREG-0313. These welds are among a total of 110 welds of 4" diameter and larger that are presently scheduled to receive augmented inservice inspection (ISI) in accordance with Section IV.B.1.b of NUREG-0313.

The 110 welds are as follows:

1. Recirculation system discharge valve bypass line - 4-4" welds (HAZ,s on weldolets).

2. Recirc. risers - 10-12" welds (HAZ's on sweepolets/ reducers).

3. Core spray - 4-12" welds (HAZ's on 316 SS flued heads).

4. RWCU - 6-4" welds (HAZ's on weldolets/flued heads).

5. Residual Heat Removal (RHR) system - 41 welds (20", 24").

6. Balance of recirculation system - 45 welds (4", 22", 28").

For No. 2 unit, PP&L intends to use the IHSI process for 103 out of a total of 105 non-conforming welds. All 105 welds are also presently scheduled for augmented ISI.

The 105 welds are as follows:

1. Recirculation system discharge valve bypass line - 4-4" welds (HAZ's on weldolets).

2. Recirc. risers - 10-12" welds (HAZ's on sweepolets/ reducers).

3. Core spray - 4-12" welds (HAZ's on 316 SS flued heads).

4. RWCU - 2-4" welds (HAZ's on weldolets).

5. Residual Heat Removal (RHR) system-40 welds (20", 24").

6. Balance of recirculation system - 45 welds (4", 22", 28").

The only other known welds that do not conform to Section III are numerous small (< 2" diameter) socket welds, used mostly on instrument piping. Because of the difficulty with reflected signal interpretation due to their small size, they are exempt from the type of inspection (ultrasonic) that is meaningful for detecting the inside-surface originating IGSCC, and therefore, PP&L believes that augmented ISI is not required for these small welds.

The SSES leak detection system has been reviewed for NUREG-0313 compliance and it is our determination that the design conforms to the requiremrnts of Section IV.B.1.a.

ATTACHMENT 2

May 1973

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.44

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

A. INTRODUCTION

General Design Criteria 1 and 4 of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," require that components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that they be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accident conditions. Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires that measures be established to assure materials control and control of special processes such as welding and heat treating and to assure performance of reliable testing programs. This guide describes acceptable methods of implementing the above requirements with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Control of the application and processing of stainless steel to avoid severe sensitization is needed to diminish the numerous occurrences of stress corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data demonstrate that sensitized stainless steel is significantly more susceptible to stress corrosion cracking than non-sensitized (solution heat treated) stainless steel. Of specific concern in this guide are the unstabilized austenitic stainless steels, which include American Iron and Steel Institute (AISI) types 304 and 316 normally used for components of the reactor coolant system and other safety-related systems. This guide does not cover stabilized stainless steels (e.g., AISI types 321 and 347) which also provide some protection against sensitization, since these materials are not currently being selected for use in light-water-cooled reactors.

Process controls should be exercised during all stages of component manufacturing and reactor construction to minimize exposure of stainless steel to contaminants that could lead to stress corrosion cracking. Since some degree of material contamination is inevitable during these operations, halogens and halogen bearing compounds (e.g., die lubricants, marking compounds and masking tape) should be avoided to the degree practical.

All cleaning solutions, processing compounds, degreasing agents, and other foreign materials should be completely removed at any stage of processing prior to any elevated temperature treatment and prior to hydrotests. Reasonable care should be taken to keep (1) fabrication and construction areas clean, (2) components protected and dry during storage and shipment, and (3) all crevices and small openings protected against contamination. Pickling of sensitized stainless steel should be avoided. Special precautions should be taken to avoid surface contamination with fluorides from welding rod coatings and fluxes. The quality of water used for final cleaning or flushing of finished surfaces during installation should be in accordance with Regulatory Guide 1.37.

Solution heat treating and testing should normally be performed on starting material. However, in order to assure the proper solution heat-treated condition of the surface areas of finished components, it may be preferable to perform the solution heat treating and testing operation at a later stage of component manufacturing.

Solution heat treating should include cooling rates sufficiently rapid to prevent precipitation of carbides to a degree that the material is not susceptible to intergranular stress corrosion. Water quenching (used for simple shapes such as bars and plates) should produce an acceptable cooling rate. However, cooling by means other than water quenching is acceptable only when the cooling rate is sufficiently rapid to prevent sensitization. This determination is made by subjecting the material to a suitable intergranular corrosion test such as Practice E of American Society for Testing and Materials (ASTM) A 262-70.[1]

Practice E of ASTM A 262-70, "Copper–Copper Sulfate–Sulfuric Acid Test," and the accompanying screening test Practice A, "Oxalic Acid Etch Test," are considered suitable tests for verifying non-susceptibility of the material to intergranular stress corrosion. Although these accelerated tests use different environments than anticipated in reactors and do not provide information relating directly to susceptibility to stress corrosion cracking in reactor environments, these tests do readily detect the presence of significant sensitization of the material, a condition which has been related with actual intergranular stress corrosion attack in reactor environments. These specific tests are identified here because they are the only known tests endorsed by a consensus standard that includes acceptance criteria (acceptable–non-acceptable basis) for the material being tested. Alternate test methods that can be qualifed are also acceptable.

Specimens for the intergranular corrosion tests from material with carbon content greater than 0.03 percent should be tested in the solution heat-treated condition. Specimens from material with carbon content of 0.03 percent or less should be tested after a sensitizing treatment of one hour at 1250°F ± 25°F.

Controls should be maintained on the chemistry of the reactor coolant and auxiliary systems fluids to which the material is exposed. Chloride and fluoride ion concentrations should be specified to be less than 0.15 parts per million (ppm) at all times. Dissolved oxygen concentrations should be maintained below 0.10 ppm during periods when the material is at elevated temperatures. When the oxygen content regularly exceeds this level, such as occurs in BWR reactor coolants during normal operation, sensitization of material that is welded without subsequent solution heat treatment should be further controlled by limiting the carbon level in the material to 0.03 percent. Carbon level control is not needed for weld metal and castings with duplex structures since these product forms with normal carbon levels have demonstrated adequate resistance to intergranular attack. Carbon level control may not be required for piping if its diameters are sufficiently small (e.g., instrument lines and control rod drive hydraulic systems) so that it could withstand a single failure without an accompanying loss-of-coolant accident as defined in 10 CFR Part 50, Appendix A.

Stainless steel subjected to sensitizing temperatures (800 to 1500°F) during fabrication (except during welding) should be retested with a suitable intergranular corrosion test (such as ASTM A 262-70) to demonstrate that the thermal treatment did not result in undue sensitization. Specimens for the retest should be subjected to a thermal treatment that duplicates the temperatures, number of cycles, holding time at each cycle, and minimum heating and cooling rate in the 800 to 1500°F range. If more than one cycle at only one temperature is to be used in production, one cycle with a holding time equivalent to the total time would be acceptable for testing purposes.

Under certain conditions material subjected to sensitizing temperatures (800 to 1500°F) during special processing may be acceptable for intended use (e.g., nitrided control rod drive material). These conditions should include, as a minimum, assurance that:

1. The process is properly qualified and controlled to develop a consistent and uniform product, irrespective of heat of material and equipment used; and
2. Adequate documentation exists that the processed material will not develop intergranular stress corrosion during its service life. Adequate documentation should include actual service experience and/or test data in simulated environments and operating conditions. Service experience should include positive evidence through destructive examination that intergranular stress corrosion did not occur.

All welding processes will result in some carbide precipitation in the weld metal and in base metal heat-affected zone of stainless steel welds, but significant sensitization does not normally result when typical welding procedures and material chemistry are used and when no further heating of material occurs. However, there is evidence that atypical welding methods using very high heat input could result in stress corrosion cracking in the heat-affected zone of the weld. To avoid this, the welding procedures and material chemistry (if necessary) should be controlled to prevent undue sensitization of the heat-affected zones of the weldments. Controls to prevent sensitization of the material during welding may include: (1) avoiding welding practices that result in the generation of high heat, (2) maintaining low heat input by controlling current, voltage, and travel speed, (3) limiting interpass temperature, (4) using stringer bead techniques and avoiding excessive weaving, and (5) limiting the carbon level of the material where section thickness makes the material more prone to sensitization.

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[1] ASTM Standard A 262-70, "Recommended Practices for Detecting Susceptability to Intergranular Attack in Stainless Steels," may be obtained from ASTM, 1916 Race Street, Philadelphia, Pennsylvania 19103.

In addition, welding procedures[2] should be qualified by passing a suitable intergranular corrosion test in all cases where the procedure is used for welding stainless steel having a carbon level greater than 0.03 percent. The qualification test should be performed using base material having the maximum carbon content anticipated and the minimum and maximum thicknesses anticipated.

As a minimum, the variables tthat should be controlled in the qualification test are heat input, interpass temperature, and welding techniques for specific section thicknesses.

C. REGULATORY POSITION

Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the following:

1. Material should be suitably cleaned and suitably protected against contaminants capable of causing stress corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of components and systems.

2. Material from which components and systems are to be fabricated should be solution heat treated[3] to produce a non-sensitized condition in the material.

3. Non-sensitization of the material[4] should be verified using ASTM A 262-70, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steel," Practices A or E, or another method that can be demonstrated to show non-sensitization in austenitic stainless steel. Test Specimens should be selected from material subjected to each different heat treatment practice and from each heat.

4. Material subjected to sensitizing temperature in the range of 800 to 1500°F, subsequent to solution heat treating in accordance with subparagraph C.2. above and testing in accordance with subparagraph C.3. above, should be L Grade material; that is, it should not have a carbon content greater than 0.03 percent. Exceptions are:

(a) Material exposed to reactor coolant which has a controlled concentration of less than 0.10 ppm dissolved oxygen at all temperatures above 200°F during normal operation; or

(b) Material in the form of castings or weld metal with a ferrite content of at least 5 percent; or

(c) Piping in the solution annealed condition whose exposure to temperatures in the range of 800 to 1500°F has been limited to welding operations, provided it is of sufficiently small diameter so that in the event of a credible postulated failure of the piping during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

5. Material subjected to sensitizing temperatures in the range of 800 to 1500°F during heat treating or processing other than welding, subsequent to solution heat treating in accordance with subparagraph C.2. above, and testing in accordance with subparagraph C.3. above, should be retested in accordance with subparagraph C.3. above, to demonstrate that is not susceptible to intergranular attack, except that retest is not required for:

(a) Cast metal or weld metal with a ferrite content of 5 percent or more; or

(b) Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800 to 1500°F for less than one hour; or

(c) Material exposed to special processing, provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

Specimens for the above retest should be taken from each heat of material and should be subjected to a thermal treatment that is representative of the anticipated thermal conditions that the production material will undergo.

6. Welding practices and, if necessary, material composition should be controlled to avoid excessive sensitization of base metal heat-affected zones of weldments. An intergranular corrosion test, such as specified in subparagraph C.3. above, should be performed for each welding procedure to be used for welding material having a carbon content of greater than 0.03 percent.

[2] Welding procedure means procedures qualified in accordance to the rules of Section IX of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

[3] Solution heat treated means heating to a suitable temperature, holding at that temperature long enough to cause all carbides to enter into solution, and then cooling rapidly enough to keep the carbon in solution.

[4] Material of product forms with simple shapes not subject to distortion during heat treatment such as plate, sheet, bars, pipe, and tubes need not be tested provided the solution heat treatment is followed by water quenching.

ATTACHMENT 3

3B-1.44 Control of the Use of Sensitized Stainless Steel (5/73)

Control of the use of sensitized stainless steel complies with Regulatory Guide 1.44, with the following modification:

1. Regulatory Position C.6: In lieu of the intergranular corrosion test specified in paragraph C.3, an alternate test method may be used to demonstrate acceptable levels of sensitization for welding procedures. The ASTM A708-74 standard is used to perform intergranular corrosion test in paragraph C.6 except that the radius of the bend test specimen is as specified in ASME Code, Section IX with weld-base metal interface located at the centerline of the bend.

2. For the NSSS components, conformance to this guide is in accordance with the generic BWR position as documented in GESSAR (AEC Docket No. STN-50-447) Chapter 5.

Reference Sections 4.5, 5.2.3, and 5.2.5

have [illegible] instances of SCC in furnace sensitized stainless steel. [illegible] there have been no instances of SCC of 304 stainless steels which were [illegible] with presently implemented controls.

Projection of the early SCC incidents to current BWRs is not justified. This position is supported by the following discussion.

5.2.3.2.1.1 Process Controls for 304 Stainless Steel. Improvements in technology fostered by extensive GE-APED research and development since the early GE BWR's have resulted in implementation of the following major processing controls for 304 stainless steel.

1) Furnace sensitized components are prohibited.

2) Welding heat input is restricted to 110,000 joules/in. and a maximum interpass temperature of 350°F is required.

3) Block welding is prohibited.

4) Restrictions are placed on cold work.

5) Fabrication and cleaning controls are specified to minimize contaminants.

6) Pickling of welded stainless steel is prohibited.

The effectiveness of these controls has been well demonstrated, by absence of a single stress corrosion cracking incident in "normal" BWR service, in the 5 years since they were implemented. (Note: "normal" is used to distinguish from abnormal service such as chloride intrusion.)

5.2.3.2.1.2 GE BWR Service Experience

1) General. All known incidents of SCC in GE BWR service have been investigated and documented. The SCC incidents are generally well known and, have been summarized in the open literature (References 3 and 4).

By far the majority of the stress corrosion cracking incidents which have occurred with welded components in BWR service occurred at the first commercial BWR constructed in the United States.

It must be recognized that the first plants were designed and constructed with the best technology available at the time. However, the state of the art did not approach the present-day technology since the present processing controls were being developed at that time. The importance of the controls outlined in Subsection 5.2.3.2.1.1 were not well recognized. Nevertheless, the overall service experience with 304 stainless steel in this first plant and in the early plants has been excellent.

080274

.2.3.3.4 Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes (Continued)

Electrodes are distributed from sealed containers or ovens as required. At the end of each work shift, unused electrodes are returned to the storage ovens. Electrodes which are damaged, wet, or contaminated are discarded. If any electrodes are inadvertently left out of the ovens for more than one shift, they are discarded or reconditioned in accordance with manufacturer instructions.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

Regulatory Guide 1.44 addresses 10CFR50, Appendix A, GDCs 1 and 4, and Appendix B, requirements to control the application and processing of stainless steel to avoid severe sensitization that could lead to stress/corrosion cracking.

All austenitic stainless steel was purchased in the solution-heat-treated condition in accordance with applicable ASME and ASTM specifications.

Cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization. Nonsensitization was verified using ASTM A262, Practice A or E, methods.

Material changes have been made to minimize the possibility of intergranular stress/corrosion cracking (IGSCC). All wrought austenitic stainless steel in the reactor coolant pressure boundary was changed to low carbon Type 316 or 304L with 0.03%

maximum carbon content. There is no piping which is service sensitive or nonconforming as defined in NUREG-0313.

For manual welds with the gas tungsten arc (GTAW) and shielded metal arc (SMAW) welding processes, the heat input was limited by weaving and welding technique restrictions. Non-weaving (stringer bead) techniques were used where possible. When required, weaving was controlled to meet the following bead width limits: for GTAW, the lesser of five times the filler wire diameter or 7/16 inch; for SMAW, the lesser of four times the electrode core wire diameter or 5/8 inch. For machine, automatic, and manual welding with processes except GTAW and SMAW, heat input was restricted to 50,000 joules per inch. Interpass temperature was restricted to 350°F for all stainless steel welds. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800°F by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization and to comply with Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel.

For commitment, revision, and scope, see Section 1.8.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components was avoided by

ATTACHMENT

NUREG-0420

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1981

NUREG-0420 SER RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1 APRIL 1981

We conclude that, with the exceptions noted above, the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions as set forth in Standard Review Plan Section 4.4 and is, therefore, acceptable.

4.5 Reactor Materials

4.5.1 Control Rod System Structural Materials

The mechanical properties of structural materials selected by the applicant for the control rod system components of Shoreham that are exposed to the reactor coolant satisfy the criteria of Appendix I of Section III of the American Society of Mechanical Engineers Code and Parts A, B, and C of Section II of the Code, and conform with our position as stated in Section 3.5.1 of the Standard Review Plan that the yield strength of cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch.

The controls imposed upon the austenitic stainless steel of the mechanisms satisfy the intent of the recommendations of Regulatory Guide 1.31, "Controls of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the component. The compatibility of all materials used in the control rod system, in contact with the reactor coolant, satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with our positions as stated in Section 4.5.1 of the Standard Review Plan. Cleaning and cleanliness control are in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

Conformance with the codes, standards, and regulatory guides indicated above, conformance with our positions on the allowable maximum yield strength of cold worked austenitic stainless steel, and the tempering or aging temperatures of martensitic and precipitation-hardened stainless steel, constitute an acceptable basis for meeting in part the requirements of General Design Criterion 26.

4.5.2 Reactor Internal and Core Support Materials

The applicant has met the requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50 by assuring that the design, fabrication, and testing of the materials used in the reactor internals and core support structure are of high quality standards and adequate for structural integrity. The controls imposed upon components constructed of austenitic stainless steel satisfy the intent of the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The materials used for the construction of components of the reactor internals and core support structures have been identified by specification and found to be in conformance with the requirements of NUREG-2000 of Section III and Parts A, B, and C of Section II fo the ASME Code. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and core support structure will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component integrity.

The material selection, fabrication practices, examination and testing procedures, and control practice performed in accordance to those recommendations provide reasonable assurance that the materials used for the reactor internals and core support structure will be in a metallurgical condition to preclude service deterioration. Conformance with the requirements of the ASME Code and the intent of the recommendations of the regulatory guides constitutes an acceptable basis for meeting the requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50.

4.6 Functional Design of Reactivity Control Systems

4.6.1 General

The control rod drive system and recirculation flow control system are designed for reactivity control during power operation. Reactivity is controlled in the event of fast transients by automatic rod insertion. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system which pumps a solution of sodium pentaborate into the primary system.

4.6.2 Control Rod System

Each control rod is moved by a separate hydraulic control unit. A supply pump provides the hydraulic control units with water from the condensate storage tank for cooling the rods and for moving them into and out of the core, with a spare pump on standby. The pump also provides water to a scram accumulator in each hydraulic control unit to maintain the desired water inventory. When necessary, the accumulator forces water into the drive system to scram the control rod connected to that hydraulic control unit. At lower pressures the volume of water in the scram accumulator is sufficient to scram the rod. At higher pressures most of the water to scram is provided from the reactor vessel. A single failure in a hydraulic control unit would result in the failure of only one rod. In addition, any single component may be removed from the control rod drive (CRD) system without disabling the protective system. The protection system has been designed to permit periodic functional testing during power operation with the capability to test individual scram channels and motion of individual control rods independently, thus complying with the requirements of Criterion 21 of the General Design Criteria.

Preoperational tests of the control rod drive hydraulic system will be conducted to determine operability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions in order to determine compliance with applicable technical specifications. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. Operable control rods are tested for com-

Information to be reported will include all abnormalities ranging from minor wear observed during normal inspection to complete failures, including failure to open or close and inadvertent operation. We will require the applicant to participate in this program.

To reduce the effects of safety-relief valve discharge to the suppression pool, the applicant has changed the safety-relief valve discharge device from a ramshead to a quencher design. The applicant has stated that the overpressure protection will not be affected by this change. From a transient standpoint, the safety-relief valve discharge critical flow is the flow of interest. The applicant has stated that the change from a ramshead to a quencher does not affect the critical flow and, therefore, does not affect the overpressure calculations. On this basis, we find the change is acceptable with regard to the overpressure protection function. In addition, a startup test will be performed to demonstrate expected safety-relief valve discharge flow.

In summary we have reviewed the system design to prevent overpressurization of the reactor coolant system. We conclude that this system, conforms to the requirements of General Design Criterion 15 and the American Society of Mechanical Engineers Boiler and Pressure Vessel code and is acceptable. However, this evaluation is subject to confirmation by the ODYN re-analyses discussed above.

5.2.6 Reactor Coolant Pressure Boundary Materials

5.2.6.1 Material Specifications and Compatibility with Reactor Coolant

The materials used for construction of components of the reactor coolant pressure boundary, including the reactor vessel and its appurtenances, have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

General corrosion of all materials except carbon and low alloy steel will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces of carbon and low alloy steel in accordance with the requirements of ection III of the ASME Code. The external nonmetallic insulation to be used on austenitic stainless steel components conforms with the requirements of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled and the proposed maximum contaminant levels have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and provide reasonably assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of structural integrity of a component.

The instrumentation and sampling provisions recommended in Regulatory Guide 1.56 for monitoring reactor coolant water chemistry provide adequate capability to detect changes on a timely basis and to effect corrective actions to bring the coolant chemistry within limits which will prevent stress corrosion.

The use of materials of proven performance and the conformance with the recommendations of the regulatory guides mentioned above constitutes an acceptable basis for satisfying the requirements of Criteria 14 and 31 of the General Design Criteria.

5.2.6.2 Stainless Steel Pipe Cracking

In September 1974, cracking was experienced in the stainless steel piping at Dresden Nuclear Power Station, Unit No. 2. This was the first of a series of incidents of intergranular stress corrosion cracking that occurred in weld heat-affected zones in Type 304 stainless steel recirculation system bypass piping systems and core spray lines. As a result of these incidents, a special task group within the NRC was formed to investigate the causes of the cracking. The results and conclusions of the task group are given in the staff technical report, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG-75/067, October 1975.

The task group found that austenitic stainless steel piping in the reactor coolant pressure boundary of boiling water reactors is susceptible to stress corrosion cracking. This was due to the pressure of oxygen in the coolant, high residual stresses, and some sensitization of metal adjacent to welds. They found that cracks were expected to be present in the heat-affected zones adjacent to welds and cracks should not occur outside these zones where sensitization has not taken place, provided the pipe material is properly annealed.

The applicant is implementing a number of the task group's recommendations for identified areas of high susceptibility. The two bypass lines have been entirely removed from the recirculation system and the existing weldolets will be capped utilizing a corrosion resistant (304L) weld inlay in contact with the coolant. The core spray safe-ends and the stainless steel spool transition pieces will be replaced with new hardware fabricated from low allow steel. The control rod drive hydraulic return line will be removed and the existing reactor pressure vessel return nozzle will be capped.

Although the applicant has taken corrective actions with regard to intergranular stress corrosion as indicated above, additional actions may be appropriate. We have issued a generic letter to the applicant transmitting NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." We will provide our evaluation of the applicant's response in a supplement to this report.

5.2.6.3 Fabrication and Processing of Ferritic Materials

Fracture toughness of the ferritic materials in the reactor coolant pressure boundary is discussed in Section 5.3.1 of this report. Welding of all components of ferritic steels was performed in accordance with the provisions of the ASME Code, Sections III and IX. This compliance with the Code provides reasonable assurance that cracking of components made from ferritic steels will not occur during fabrication.

lding procedures used for ferritic steels in limited access areas com..y he intent of Regulatory Guide 1.71, "Welder Qualification for Areas of d Accessibility." The fabrication practices and examination procedures med in accordance with these recommendations provide reasonable assurance ferritic stainless steels in the reactor coolant pressure boundary will tisfactory in locations of restricted accessibility.

ormance with the regulatory guides mentioned above constitutes an acceptable s for meeting the requirements of General Design Criteria 1 and 14.

6.4 Fabrication and Processing of Austenitic Stainless Steel

hin the reactor coolant pressure boundary, no components of austenitic stain-s steel have a yield strength exceeding 90,000 pounds per square inch, in cordance with our position as stated in Section 5.2.3 of the Standard Review an.

e controls imposed upon components constructed of austenitic stainless steel sed in the reactor coolant pressure boundary and for the reactor vessel and ts appurtenances satisfy the intent of the recommendations of Branch Technical osition MTEB 5-1 on Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (Microfissures) and in a mettalurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with the regulatory guides mentioned above constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

5.2.7 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR Part 50, requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess structural and leak tight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected prior to operation and periodically during service at Shoreham. The design of the ASME Code Class 1 and Class 2 components of the reactor coolant pressure boundary in Shoreham incorporates provisions for access for inservice inspection in accordance with Section XI of the ASME Code. Methods have been developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

Section 50.55a(g), 10 CFR Part 50, defines the detailed requirements for the preservice and inservice inspection programs for light water cooled nuclear power facility components including supports. Based upon a construction permit

6 ENGINEERED SAFETY FEATURES

The purpose of the various engineered safety features in a nuclear power plant is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. Systems and components designated as engineered safety features are designed to be capable of performing their function of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They are designed to seismic Category I standards and they will function even with complete loss of offsite power. Components and systems are provided with sufficient redundancy so that a single failure of any component or system will not result in the loss of the plant's capability to achieve and maintain a safe shutdown of the reactor. The instrumentation and control system for each engineered safety feature is designed to the same seismic, redundancy, and quality requirements as the system it serves. Instrumentation and control systems are discussed in Section 7 of this report.

6.1 Engineered Safety Features Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I to Section III of the American Society of Mechanical Engineers (ASME) Code, or Parts A, B, and C of Section II of the ASME Code. The controls imposed on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of our position on Regulatory Guide 1.31, "Control of the Use of Sensitized Stainless Steel."

The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with the ASME Code and regulatory guides mentioned above, and with our positions on stainless steel, constitute an acceptable basis for meeting applicable requirements of Criteria 35, 38, and 41 of the General Design Criteria and is acceptable.

We have evaluated the protective coating system used inside the primary containment and determined that the system meets the recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." The total amount of unqualified paints and organic materials inside the primary containment is estimated to be negligible. We therefore, conclude that the applicant's protective coating system is acceptable.

6.2 Containment Systems

The containment systems for the Shoreham Nuclear Station include a Mark II type containment structure as the primary containment, a secondary containment surrounding the primary containment and housing equipment essential to safe

REGULATORY GUIDE 1.44

Initial Issue: Revision 0, May 1973

Current Issue: Revision 0, May 1973

La Salle C.P. Issued: September 10, 1973

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Regulatory Guide 1.44 describes acceptable methods for the control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking.

The purpose of this guide is to address 10 CFR 50 Appendix A, GDC's 1 and 4, and Appendix B requirements to control "the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking." The guide proposes that this should be done by limiting sensitization due to welding as measured by ASTM A262 Practice A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steels.

Tests by General Electric indicate that the test specified by A262 A or E (Detecting Susceptibility to Intergranular Attack in Stainless Steel) detects sensitization in a gross way, and that the tests do not provide a precise method of predicting susceptibility to stress corrosion cracking in the BWR environment.

All austenitic stainless steel for LSCS Units 1 and 2 was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Carbon content was limited to 0.08% maximum, and cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization.

Welding heat input was restricted to 110,000 joules per inch maximum, and interpass temperature to 350° F. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite. As deposited austenitic welds were controlled to have at least 3% ferrite content. See Subsection 5.2.3.4 for specific reference to control of the use of sensitized stainless steel. Additionally, grinding of field erection welds was prohibited on the reactor fluid side of Class 1 stainless steel pressure boundary pipe.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800° F by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization and to comply with the intent of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

We believe that we comply with the intent of this guide via incorporation of the alternate approach cited above.

ATTACHMENT C

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Light Blue Volume
General Info. on NPE
P.1

GENERAL INFORMATION ON NUCLEAR POWER EXPERIENCE (NPE)

NPE compiles and reports on the operating experiences of all large light water nuclear power plants in the U.S.A. We report experiences of LWR's located in other countries also, but so far not as much information is available on those plants.

We concentrate on operating problems (equipment breakdowns, malfunctions, outages, etc.). Because of this our output tends to take on a very negative slant. Readers should keep in mind that the information we supply is not indicative of overall plant experience. One has only to glance through our "Availability" section to find that, on the whole, nuclear power plants are compiling envious records of reliable service.

Purposes of NPE

To help our subscribers:

- stay current with the problems occurring in the nuclear power industry,
- and to assist them in retrieving information on past problems.

By doing this, we expect to help our subscribers

- prevent outages,
- and save engineering and technician manhours.

Our System

Each month we systematically review a very large amount of literature (periodicals, technical papers, technical reports, correspondence between the plant owner and the NRC, etc.). During this review, we select only the information pertaining to operating problems. This information is condensed somewhat (but we try not to lose content) and categorized. At the end of the month we send the information to our subscribers and they file it in the loose leaf filing system which we supply.

Problem Descriptions

In writing the descriptions of the operating problems (filed in volumes BWR-2 and PWR-2) we try to include (if the information is available):

- where the problem occurred,
- when it happened (if it was prior to the start of commercial operations, the appropriate phase of startup testing, e.g., "during fuel loading"),
- what happened,
- why,
- how it could have been prevented,
- and how it was corrected.

References

At the end of each problem description, there will be one or more lower case letters enclosed by parentheses. These letters direct the reader to the reference sections located at the back of either volume, BWR-2 or PWR-2, and indicate the source(s) of the information.

The Loose Leaf Filing System

There are 5 basic volumes (some have more than one binder):

- A thin Light Blue Volume is used for a number of miscellaneous purposes. It contains a key word index, an availability summary, a list of abbreviations and filing instructions. In addition there is a section entitled "LATEST MONTH OPERATING EXPERIENCES" that can be used (if the subscriber wishes to do so) to file the latest information for volumes BWR-2 and PWR-2 for one month. In this way readers can use any spare time they have during the month to look over the new material. At the end of the month the pages are filed in volumes BWR-2 and PWR-2.

  Note: New subscribers should decide if they will use the "LATEST MONTH OPERATING EXPERIENCES" feature and instruct the person(s) responsible for the filing, accordingly.

- Volume BWR-1 (red) is indexed by boiling water reactor plant naem. It serves as a companion volume to the more important volume BWR-2. Volume BWR-1 contains brief plant descriptions and operating histories for each plant. The descriptions are prepared and issued at roughly the same time that a new plant "starts up" and becomes commercially available. The initial versions of the histories are issued after the plant has acquired significant experience.

- Volume BWR-2 (red) is indexed by boiling water reactor plant system and/or component. This is the most significant part of the NPE system (for BWR's) and is where the information on the operating problem is filed. This volume also contains cross-references (plant name vs. problem item number.).

- Volume PWR-1 (dark blue) is the pressurized water reactor equivalent of volume BWR-1.

- Volume PWR-2 (dark blue) is the pressurized water reactor equivalent of volume BWR-2.

Plants Already in Operation

Basically we will be reporting the experiences uncovered during the previous month. However, about 2 dozen plants started operating before we did (May 72) and we felt that our compilation would be incomplete if we did not go back and report on those old experiences. We did this during our first year of existence.

Staying Current

We publish a large quantity of information every month. Because we realize that most people do not have much spare reading time we have designed a number of "reading time savers" into our system:

- the two column format for easier reading,

- an underlined upper case headline for each operating problem (you can skip the details of problems you are not interested in),

- basic filing is by plant component or system. (if you are only interested in instrumentation and control, you can skip everything else; the page numbering system provides the clues),

- the page numbering system (at the top of each page) always tells you what type of reactor (BWR or PWR) and what component and/or system you are reading about,

- the "N" (for New), "O" (for Old) and "R" (for revised) code numbers in the margins (you may only be interested in new or revised information).

Retrieval of Information

There are many reasons why a person may wish to enter the loose leaf filing system to retrieve an experience. For example, you may be experiencing a problem and wish to know how someone else has solved the same problem in the past. NPE has 3 features that allow quick entry into the filing system:

- The basic method of filing is by BWR or PWR plant system and/or component.

- The key word index groups all problems with the same key words.

- If you know that a particular plant had a particular problem, refer to the cross-reference for that plant.

Note: Some problems may still be filed in the "Latest Month Operating Experiences" section.

Abbreviations

To save space and reading time we use a number of abbreviations. The "Light Blue Volume" contains our abbreviation list. In constructing our list we started with the American National Standards Institute abbreviations (ANSI Y10.19 - 1969). We had to modify some of them so that they agreed with what we thought was more common practice and of course we had to add some to the list.

Page Numbering System

The "Filing Instructions" in the "Light Blue Volume" contain an explanation of our page numbering system. Basically what we are attempting to do is to keep the reader informed as to exactly what section he is in. This is especially helpful if the page is temporarily stored in the "Latest Month Operating Experiences" section.

Issue Date

The last month in which a particular page was issued or revised is always shown in the lower right hand corner of the page.

Code Letters in the Margins

We use the following code letters in the left hand margins of the pages in volumes BWR-2 and PWR-2.

N - New - - - - A problem that we learned about for the first time during the past
N month.

O - Old - - - - A problem that we are including for the first time but only as
O a result of our efforts to thoroughly and/or completely report on significant problems.

R - Revised - - - - We may have uncovered some additional information during the
R past month that concerns a previously reported problem, or we may be correcting a mistake.

Feedback

NPE encourages feedback from subscribers (recommendations, suggested improvements, experiences we may have missed, etc.) and will willingly act as an information processing point for the industry.

1. FIRE INSIDE VESSEL

Quad-Cities 1 - Sept 70 (construction)

A smoldering wood fire occurred on top of the lower core plate. The fire started from the heat released from a 1500 W quartz flood light which was mounted above the temporary wood decking used on the lower core plate. This flood light ignited either the insulation material on a welding cable coiled on the wood decking or the plywood material of the decking. The fire spread across a polyethylene plastic covered wooden frame which was utilized during erection operations to control air currents around an optical alignment instrument.

Materials consumed included about sixteen 1 ft^2 core plate covers, 40 ft of neoprene welding cable, 3 ft of plastic ventilation air duct and 80 ft^2 of polyethylene.

Three unsuccessful attempts to extinguish the fire were made from below the core plate with CO_2 and dry chemicals. A fourth attempt from above, with demineralized water was successful.

The fire left a deposit of black soot on the interior surface of the vessel shroud in the area above the fire. Also, the bottom of the vessel contained a small amount of water and bits and pieces of debris. On the lower core structure, black soot and combustion products were deposited on the exposed metal surfaces.

The interior of the RPV was cleaned with demineralized water, acetone, demineralized water with 500 ppm trisodium phosphate, wire brushes and putty knives.

Analysis of the occurrence, the debris, and measurements indicated that the temperatures did not sensitize the lower core plate structure.

Because the dry chemical extinguisher contained potassium chloride, they performed another cleaning operation of the vessel interior before performing the system hydro.

Corrective measures included 1) fire retardant paint on wood on the lower core plate, 2) discontinuing the use of quartz lamps, and 3) additional approved fire extinguishers located in the area and in the vessel. (cl)

2. WELDING OUTFIT FOUND IN JET PUMP RISER

R Quad-Cities 2- Apr 72 (startup testing)

Jet pump flow comparison checks were being performed as a final check prior to installation of the steam separator, steam dryer, and reactor vessel head. This test requires that both reactor recirculation pumps be running at the same speed with the recirculation header crosstie valves closed. The indicated flow rates on 2 of the jet pumps were about 50% of the other 18 jet pumps. One jet pump was removed and the obstruction was found to be a heliarc welding outfit consisting of:

- A [illegible] in. OD rubber hose, 24½ ft long, complete with fittings attached.
- A 3/8 in. OD rubber hose, 12 ft long, complete with "Chicago" fitting on one end and a copper fitting on the other.
- A 2 conductor welding cable, 24½ ft long, attached to a TIG welding gun. These conductors were covered with asbestos insulation which had been repaired with plastic tape.

The outfit was fished out using "J" hooks. The riser was inspected and the jet pump was reinstalled into the reactor vessel.

R The test program was delayed about 4 days. (ej,ir,so)

3. GUIDE TUBES CAME LOOSE - CAUSED FUEL DAMAGE

Tarapur 1 - Aug 71

Both reactors at Tarapur are operating with only 284 instead of their ultimate capability of 368 fuel assemblies. Therefore 20 peripheral control rod guide tube positions have less than the full complement of 4 fuel bundles. The fuel support positions which are unoccupied by fuel are plugged with fuel support plugs. The fuel supports are attached to the guide tubes by welds. No control rods or drives are installed in these 20 locations. To compensate for the lack of the normal downward force afforded by the fuel weight, the CRD thermal sleeves are restrained by a hold down device. The thermal sleeve is engaged with the guide tube bayonet and then locked against rotation with a set screw which engages a slot in the CRD housing flange. A core support pin which engages lugs on the guide tube, prevents guide tube rotation.

The set screw which is installed after the thermal sleeve is positioned prevents the thermal sleeve from rotating. The set screw is turned into the thermal sleeve ring until the cup point bottoms in the CRD housing slot. It is backed out ½ turn to prevent shearing during relative thermal movement between the CRD housing and the thermal sleeve. The set screw is not locked. For CRD housings in which no CRD's are installed nothing prevents the screw from backing out.

During a refueling outage, damage was discovered. Two of the 20 guide tube assemblies had become disconnected. Each of these assemblies have 2 fuel bundles installed. Once disconnected these assemblies were lifted during reactor operation by the hydraulic forces created by incoming reactor water flow. They rose a distance of ~3 ft until restrained by the underside of the shroud head steam separator assembly located above the core. Calculations confirm that if not restrained, it is possible for guide tube assemblies containing 2 fuel bundles to rise and fall as a function of core flow.

The motion of these 2 components caused damage to themselves and to adjacent structural components. Holes were worn through the fuel channels of 2 neighboring fuel elements and a small section of the cladding of 3 fuel rods was also worn through. In addition, the motion of the lifted control rod guide tubes broke 2 in-core guide tube stabilizer rods. All damage was confined to the immediate vicinity of the 2 lifted guide tubes.

Unlocking of the 2 assemblies appears to have been caused by a failure of the set screws due to improper installation and/or fatigue failure. Lifting of the 2 guide tubes was caused by the eventual random rotation of the thermal sleeves and the subsequent unlocking of the guide tube combined with a lack of sufficient downward force to withstand the upward hydraulic forces on the guide tubes. The upward force is due to the pressure drop occurring across the core plate. It has been calculated that installing 2 or more fuel bundles in each assembly provides a sufficient downward force to overcome any lifting forces which are present during normal operations.

The design of plants in the U.S. differs significantly from that at Tarapur. Therefore, no potential for a comparable incident exists. (er,bcv)

4. WELDING PURGE DAM LEFT IN LPCI - BLOCKS JET PUMP

Dresden 3 - Mar 71 (after initial criticality)

During functional tests of the Recirculation System, a low flow indication was observed on a jet pump. Normal flow indications were recorded previously. Checks revealed no problems with the flow transmitter or indicators. The reactor head and vessel internals were then removed to gain access to the jet pumps. The vessel water level was lowered to ~ 2/3 core height. Radiation levels were ~ 35 mr/hr. The cause for reduced flow was found to be a welding purge dam lodged in the transition casting of the jet pump at a point where the flow is divided from the riser inlet to 2 nozzles for two jet pumps. The purge dam was lodged against the casting and blocked ~ 3/4 of the flow to the jet pump. Upon removal, the purge dam was revealed to be a standard plywood and rubber dam, 16 in. in dia, commonly used in construction heliarc welding operations. Due to the size and characteristics of the dam, it must have originated from the 16 in. LPCI discharge piping. The LPCI system is the only source of 16 in. pipe that discharges directly into the recirculation piping.

The dam consisted of a sandwich of four 3/8 in. thick plywood semi-circular segments and 2 circular 1/8 in. thick red rubber sheets held together by ¼ - 20 x 1¼ in. stove bolts. One half was broken in two, the other half was badly fouled and the 2 halves were held together by the normally used chain. Various portions of the internals were inspected, vacuumed, and flushed.

Missing from the dam and not recovered were 30 $in.^2$ of rubber in small pieces, 16 $in.^2$ of wood splinters, 4 nut screw combinations, 1 washer and 1 nut. LPCI and HPCI testable check valves were disassembled but no foreign materials were found. All fuel assemblies were removed, gamma scanned and inspected. Additional checks on various systems and components were made. (fz)

5. JET PUMP VIBRATION - MOUNTING BRACKET PROBLEM

Dresden 2 - Startup

The vibration test program showed that the jet pump riser brace experienced high vibration whenever a mismatch in pump speed between the recirculation pumps occurred. This could occur when one recirculation pump was operating at partial flow while the other one is at full flow. It could also occur following a pump trip when an attempt is made to start the inoperable pump. The hydraulic forces can cause an increase in jet pump vibration, throwing an additional load on the mounting bracket. Under such a condition half of the jet pumps would be operating slightly above their normal rating, forcing backflow through the remaining jet pumps, which have insufficient driving flow. The increased vibration occurs in the low flow pumps and occurs at the point which the flow is at the point of reversing. An interlock system was installed that:

- automatically prevents restart of an idle recirculation pump when the operating pump exceeds 65% of design speed, an alarm sounds if such a restart is attempted.
- prevents recirculation pump operation (positive interlocks on pump speed controls) in the vibration regions.
- sounds alarms if changes occur and the undesired regions are approached.

The jet pump brackets were strengthened in all plants subsequent to Dresden 2 (Millstone I, Fukushima I, etc.) The same interlocks were installed at Quad Cities 1 & 2. (bg, fr, ga)

6. POWER ASYMMETRY - CORE INLET TEMPERATURE DIFFERENCES

Dresden 2 - Spring 70 (plant startup)

Differences between TIP and LPRM readings indicated that a power asymmetry problem existed. The asymmetry ratio was as high as 1:1.16.

During a July 70 shutdown some low sensitivity LPRM's were replaced. A gamma scan confirmed the asymmetry and indicated a ratio of ~ 1:1.07.

Various tests and analyses were conducted to rule out the following as causes:

- fuel, control rod or control curtain effects.
- flow non-uniformity or flow variations between bundles.

It was concluded that the cause was non-uniform temperature of the core flow at the core inlet and that this could be corrected by properly distributing the feedwater flow around the downcomer annulus.

The feedwater sparger on Dresden 3 was modified (enlarging and plugging selected holes) and tests run in 1971 support the non-uniform temperature theory, and the idea that the situation can be corrected by properly distributing the flow around the downcomer annulus. However, as of Feb 72, the results were not considered conclusive because of errors and uncertainties in the incore instrumentation data.

Feedwater sparger modification were also made at Quad Cities 1 and 2 and other plants.

They believe that the asymmetrics do not present a safety problem. (fr, gk, gl, hk)

7. NEUTRON SOURCE ENCAPSULATION FAILURE

Big Rock Pt. - Apr 72

Two auxiliary neutron sources had been installed for about 1 yr. They are antimony-beryllium sources with an inner encapsulation of SS and an outer encapsulation of a zirconium alloy. A visual inspection of the sources revealed that the outer encapsulation had failed in the area of the lower end-cap weld on both sources. The failed area had the appearance characteristic of typical hydride attack.

The lower end cap was removed from one of the sources and the inner SS encapsulation was visually examined. No abnormalities were detected. It was concluded that the cause of the failure was due to internal contamination.

A new outer encapsulation was provided using zircalloy tubes with the upper end caps welded to the tubing. The SS encapsulation was placed in this tubing and a lower end cap inserted. Mechanical forces were used to crimp the tubing to the lower end cap. Because of the activity of the sources, it was impossible to quickly perform the repair in a moisture free environment. Flow holes were provided at the top and the bottom of the sheathing to permit flow of water and steam in the annulus to eliminate the effects of having moisture trapped inside the zircalloy tubing at the lower end cap. Because of this they planned to monitor for antimony in their water chemistry program. (iz)

8. JET PUMP HOLD DOWN ASSEMBLY FAILURE

Quad-Cities 2 - Aug 72

A recirc pump trip occurred while at 84% power. During recovery, and while at 60% power, various instrumentation showed abnormal readings which led to the conclusion that the No. 17 jet pump had failed.

The jet pump assembly provides for removal of the inlet-mixer section if inservice inspection is desired. The inlet-mixer is mechanically clamped to the riser with a remotely operable beam bolt arrangement which transmits a 25,000 lb downward clamping force. This is ~ 3x the upward force and ensures minimal leakage at the spherical sealing joint at the end of the riser. A lock welded keeper piece over the bolt prevents rotation of the bolt, thereby maintaining the beam load. A pair of lock welds are provided to prevent rotation. The lower discharge end of the mixer is held laterally by a restrainer bracket which must be opened for mixer removal. It is locked in the closed position by a pair of bolts and they also have lock welded keepers. The swinging gate portion of this bracket is latched to a fixed frame so that even if the bolts were loose the swinging member would have to be raised 0.3 in. for disengagement.

Following shutdown and head removal, the inspection revealed that pump No. 17 had lifted 3 to 6 in., rotated about 45° and was resting between No. [illegible] and the vessel wall because of a failure of the hold down assembly. The beam bolt assembly and insert washer were missing, the retainer and connecting 1/2 in. bolt (capture the beam and bolt during servicing) were detached, and one keeper on one of the lower restrainer clamp bolts was missing.

A underwater search (with TV) located and recovered all missing parts with the exception of the 2.25 in. dia insert washer. They feel that the most likely resting place would be in the vessel on the bottom of the outer annulus. During the process of raising the beam bolt retainer, it was inadvertently dropped into the opening in the support shelf left by the removed jet pump No. 17 mixer and it settled into the bottom of the reactor lower plenum. It was sighted but they were not able to recover it. Later, during repair operations, portions of a welding outfit were inadvertently dropped. All parts were recovered except for a small 1 in. OD neoprene bushing and half of its companion 1 3/8 in. OD compression cap. Analyses indicated that none of the lost items would pose a safety problem.

There was no damage to the riser ports and seat, the restrainer gate mount or the slip-joint seat of the diffuser. Both of the beam bolt keeper lock welds were broken. Underwater TV exams of the other pumps revealed:

- 3 of 38 beam keeper lock welds appeared to be marginal.
- 10 of 38 restrainer gate keeper lock welds appeared to be marginal.
- 1 of 19 restrainer gate wedges was improperly positioned.
- 1 lower restrainer clamp bolt was loose by several turns but the keeper was soundly welded in place.

A metallurgical exam was performed and a special 300 ft-lb counter-clockwise torque test was applied to all beam bolts. The keeper lock welds on 2 pumps sheared during this test indicating inadequate beam preload.

As a result of exams, analyses and tests, they concluded that the failure occurred because the hold down beam bolt keeper was inadequately field welded and/or field preload was not applied to the hold down beam. Either of these could have led to loosening and the disassembly of the pump.

The loss of the retainer and 1/2 in. bolt from the top of the mixer was probably unrelated. The bolt is lockwelded to the retainer; however, there was very little weld material. Normal vibration probably rotated the bolt out and allowed the bolt and retainer to fall free. Corrective actions (on all pumps) included:

- correcting all observed installation deficiencies,
- refurbishing pump No. 17,
- breaking all beam bolt keeper welds and reapplying required beam and beam bolt preloads,
- reweld all beam bolt keepers,
- recheck that the welds were applied to meet torque testing and that the beam bolts do not move when torqued in both directions (this also checked pretensioning), and
- reconfirm that all beam bolt keeper welds were sound after the torque test.

N The duration of the shutdown was 54 days. (mv, qv, [illegible])

9. STEEL BRACE DROPPED ON & DAMAGES TOP FUEL GUIDE

Peach Bottom 3 - May 72 (construction)

During erection of the structural steel in the reactor building, a diagonal brace (8 x 6 x 9/16 in. angle, 13 ft 10-3/4 in. long, weighing ~ 360 lb) slipped from its choker and dropped from an estimated height of 30 ft. The steel erection subcontractor was evidently following accepted rigging procedures. The angle brace pierced the shipping container of the top fuel guide. The exact extent of damage could not be determined at the time because the reactor building crane was not yet operational. In July they were able to uncrate the guide and make dimensional checks. Beams which make up the cells of the guide were bent and pushed out of position. The unit was shipped off site to a location where the special optics required for a thorough examination were available. Steps were taken to protect equipment stored in the reactor building at Elev. 234 with planking and tarpaulins. One beam was replaced. (nr,agi)

10. JET PUMP CRACKS - CARBURIZATION

Dresden 2 & 3 - early 1969 (construction)

An investigation of castings used in the original jet pump assemblies revealed cracks in the transition piece, 180° elbow, nozzle, coupling and blange. The cracks were caused by surface carburization. The casting supplier had been employing the shell-mold process and used carbonaceous materials in the binder and wash. The foundry process was modified and replacement castings were purchased and installed. (oe)

11. FW SPARGER CRACKS

Millstone 1 - Oct 72

The reactor internals were inspected as part of the investigations that were performed after the chloride intrusion incident (see VI. C. 3). Visual examinations revealed failures of the FW spargers. An extensive metallographic and analytical investigation was performed to determine the cause of the failures. It was concluded that the failures occurred prior to the chloride excursion as the result of high-cycle fatigue following a condition of inadequate cold spring. New spargers of modified design were installed with a carefully controlled procedure to ensure that proper cold spring was applied to all spargers.

There are 4 spargers located in the upper portion of the vessel. Their primary purpose is to distribute the reactor FW, although the reactor cleanup loop also discharges back through the B FW line (SE and SW spargers). A series of 1 1/2 in. holes, spaced at equal intervals along each sparger branch, discharge the water radially into the flow passing from the steam separators down into the jet pumps.

Although access to reactor internals does not require it, the spargers are designed to be removable. The thermal sleeve, welded to the sparger at assembly, is a slip-fit in the vessel nozzle. Brackets on the end of the sparger [illegible] which engage vessel mounting brackets [illegible]. Slots in the sparger bracket allow for relative motion between the vessel and sparger due to thermal expansion.

Wedges bearing against the vessel wall are used to preload the spargers and prevent vibration. The amount of spring, or preload, is enough to resist the force of the FW flow and any calculated transients without allowing lifting from the installed position. The calculated flow load for 105% rated flow (360°F) is 1316 pounds and would produce a 0.194 in. displacement. The reactor assembly drawing specifies a $5/16^{+1/16}_{-0}$ preload. The max sparger stress, at the junction of the header and Tee box, is calculated for 105% rated power at 10,970 psi and 13,580 psi for 5/16 and 3/8 in. displacement. The original spargers were built by Stearns-Rogers to GE drawings and specs.

The 4 spargers were removed from the reactor vessel and were examined on the reactor refueling floor. The moderate radiation field from the spargers prevented any extended close visual observation. All 4 spargers showed extensive cracking at the connections between tee box-to-headers and tee box-to-thermal sleeve. The E header of the NE and the N header of the NW sparger were hanging on by small ligaments at the bottom of the headers with open through-wall cracks extending around the remainder of the joint. Open cracking was visible on 3 spargers at some of the tee box joints. PT revealed additional cracking in all 4 spargers. The cracks tended to be singular at simple joint configurations such as the machined undercut on the tee box near the thermal sleeve joint, and multiple at complex joint configurations such as the rear corners of the tee box. The SE sparger showed the least amount of header-joint cracking. All cracks were on the external surfaces of the spargers except for the 2 open cracks in the NE and NW sparger.

There was no evidence of general cracking at any other point on the spargers. A few isolated cracks were detected in fillet welds on the lifting lugs and brackets, and one small crack was located near the middle of the South header of the SE sparger. Other isolated cracks were located at work-hardened areas such as sheared plate edges, drilled hole surfaces and a ding mark on the East header of the SE sparger.

The end brackets of all spargers except the SE showed semicircular wear patterns, on the vessel side edges of the slots, caused by the nitrided SS pin bearing on the annealed SS bar. In some cases octagonal wear patterns from the vessel bracket were observed. The pin keepers and the pin handles also showed wear patterns caused by high frequency vibration of the pins. One SW pin keeper was entirely broken off.

Metallographic analysis coupled with the evidence of wear and long-term vibration of the sparger components indicated that the major failure mechanism was fatigue and that the failures were transgranular in nature.

The most probable cause was improper installation. This resulted in inadequate preload, and subsequent failure by mechanical fatigue. The extensive damage and oxides indicated a considerable period of operation after the failure, which confirmed that the cracking was not a result of the chloride incident.

Wear of the therm[illegible] [illegible] nozzl[illegible] spacer button and [illegible] [illegible] by the nitrided pin both indicat[illegible] [illegible] number of cycles of relative movement. Wear of t[illegible] vessel mounting brackets and mating sparger bracket showed that these members provided the restraint.

Lack of assembly preload, together with the thermal-sleeve gap produced by the assembly procedure, was sufficient to explain the severe vibration which was evident from wear patterns on 3 of the 4 spargers.

Basically, a sparger was set in place with 1/2 in. spacers at the point where preload was applied. The brackets were adjusted to allow the pins to engage the reactor sparger support brackets and then welded. The sparger was removed and wedges, whose thickness provide for proper preload, were welded to the sparger. Hydraulic jacks were used, at final installation, to spring the sparger enough to insert the pins. Procedures called for recording of the gap, between the sparger and the vessel wall, during fit-up and at final installation, to allow preload to be checked.

A review of the installation data indicated that, if the gaps were recorded properly, the preloads did not meet the design intent. The SW sparger, Nozzel 4B, indicated a "negative" cold spring; i.e., the final gap 9/32 in. less than the inital gap. Recorded cold spring values on the other spargers varied from 1/32 to 9/16 in.

The most likely cause for improper preload was believed to be the weld distortion. When welding the cold spring wedges on the pipe, experience had shown that weld distortion tends to increase the chordal length of the sparger. This can reduce the cold spring if not corrected. The problem was first discovered at the Vermont Yankee plant when, prior to operation of the plant, the spargers were removed for modifications to improve FW distribution. It was discovered that the pins were easily removable, despite the supposed preload. It was necessary to add draw beads to restore the design preload. Newer plants with the wedge design use a template, or its equivalent, to insure that the sparger can be returned to its original shape after the wedges are welded in place. There were no draw beads on the Millstone sparger, so it was assumed that this effect was over-looked.

To expedite the sparger replacement program, the sparger design being built in GE shops for 218 in. vessels was modified to fit the 224 in. Millstone vessel. This allowed the incorporation of several design improvements made since the original Millstone design. Considerable improvement had been made in the distribution of FW to the various jet pumps. The latest criteria called for the flow to be equal $\pm 2.5\%$ to each segment associated with a jet pump. A series of tests had been run on flow distribution, and a computer program developed to obtain the correct hole patterns to achieve uniform flow.

Sparger distortion during site installation, which occurred during the welding of the preload wedges, required considerable rework in the field. This may have been the cause of inadequate preload in the original Millstone installation. The new design provides a shop-welded saddle, which mounts a [illegible] [illegible] [illegible] is allows considerable adju[illegible]ment [illegible] [illegible] at the preload point, including compen[illegible] for any weld distortion.

The new design is made using 6 in. Schedule 80 pipe (rather than Schedule 40) for the sparger legs. This allows a higher preload to be imposed with a given max stress. The tee box (divider) was reduced from 1 in. to 0.5 in. wall thickness, which provides some flexibility and reduction of stress in the attachment weld of the sparger leg.

A special computer program is used to calculate sparger stresses. These are at 12 points, mainly in the pin and bracket area, and in the tee box and attachment zone. The new installation plans call for a 0.50 ± 0.030 preload which will result in a preload of 530 lbs. Under normal operating conditions, the max stress produced by this preload is 17,460 psi at the sparger leg attachment to the tee box.

Additional specs control at least 3 problem areas in which the original sparger was weak: (1) the material is purchased in the solution annealed condition, with a max hardness of Rockwell B94; (2) the parts are not heated for forming, and must be solution-heat treated if processed in a manner inducing cold work; and (3) welding is done by the gas tungsten or gas metal-arc process with controls on heat input and ferrite content of the weld metal. This process avoids potential fluoride contamination from weld-flux fumes. In addition, the cutting fluids as well as other materials used in processing, and the cleaning process are controlled to prevent contamination by chlorides, fluorides, and other potentially harmful materials.

The new spargers were installed using a modification of the initial procedure and recording detailed measurements for subsequent engineering evaluation. The spargers were shipped to the plant site with the thermal sleeve welded in place but not cut to proper length. The thermal sleeve was inserted in place on the FW nozzle and measured in order to cut to proper length.

The in-vessel sparger support brackets were machined to provide a flat upper surface removing the portion of material that was partially worn away by vibration of the original spargers.

The sparger was again lowered into place and the thermal sleeve inserted in the nozzle. A measuring pin was utilized to determine the proper positioning of the end brackets on the sparger. This positioning applied a measured amount of preload to the sparger. The sparger was again removed from the vessel and the end brackets welded in place.

The sparger was reinstalled and the ends forced into position with a spreader bar so the pins could be installed. The gap between the sparger and the vessel was measured and increased by turning the jack bolts to provide the required final value of cold spring. The jack bolts were then tack welded in place.

Measurements were obtained as specified during the installation. (pz,qa,th)

12. CRACKS IN CORE SUPPORT GRID

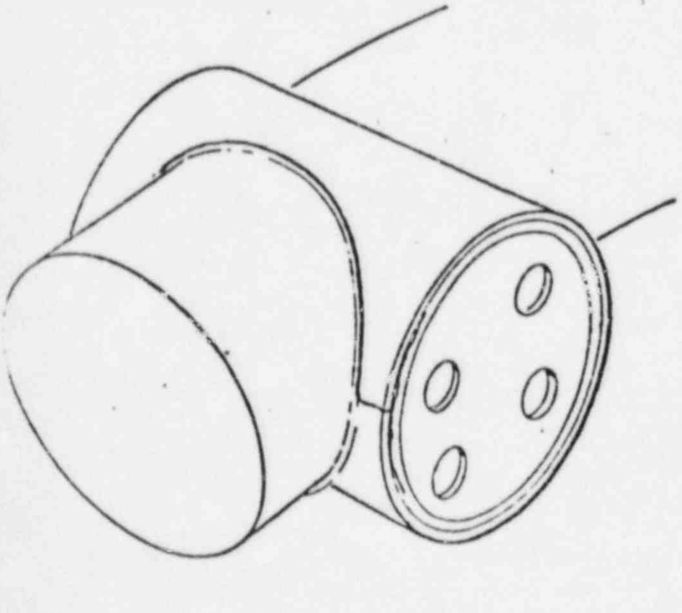
Dresden 1 - 1961 & 62

During 1961 inspections, 8 minor cracks or irregularities were found at the welds of the core support grid. Although these cracks did not affect the structural integrity of the reactor internals they monitored them to ensure that propagation had not occurred. During the refueling outage that started in Nov 62 a borescope was used to monitor 99 of the welds previously inspected, including the previously noted cracks, and 44 additional welds. They had to move 29 fuel assemblies to perform these inspections. The condition of the cracks had not changed. One more small crack was found in the additional area inspected. (ra)

13. FW SPARGER CRACKS - HIGH CYCLE FATIGUE

Millstone 1 - Apr 73

A planned shutdown and inspection (as part of the salt water intrusion recovery surveillance program, see VI.C.3) revealed new cracking in 2 of the 4 new FW spargers (see II.11). Subsequent exams disclosed that the 3rd and 4th spargers had PT crack indications. This 2nd set of spargers, No.2 design, replaced the originally installed which was found to be severely cracked. Based on the analysis which followed the chloride intrusion incident, it was concluded that the cracks in the spargers had occurred prior to the chloride transient as a result of high cycle fatigue following a condition of inadequate cold preload. The failure of the 2nd set of spargers, in the absence of chloride, substantiated the conclusion regarding the initial failure. However, it raised questions concerning the completeness of the definition of sparger operating conditions used as a basis for the Millstone design.



Teledyne, Southwest Research and GE were all involved in the investigative program.

Underwater TV revealed that both the NW and NE spargers were cracked in the junction box and front plate. Subsequent close-up examination (in vessel) revealed loose jack bolt/vessel contact, loose end bracket pins, and changes in sparger proximity to the vessel wall on all but the SE sparger. Tests indicated that only the SE sparger had maintained the cold as-installed preload (~6000 lb). The SE sparger is the entry point for clean up system return flow. While the SW, NW, and NE spargers showed a resistance to movement, none exhibited any significant preload.

Further inspection revealed:

- Some wear (~1/16 in.) in sparger pin slots and on the pins.
- Wear on top of RPV brackets which support sparger end brackets.
- Better definition of cracking on NW and NE spargers.
- A small (~1 in. length) interrupted linear PT indication on the top side of the SW sparger.
- A small (~1 7/8 in. length) interrupted linear PT indication on the bottom side of the SE sparger.
- Thermal sleeve/nozzle support pad wear.
- Broken tack welds on the 2 lower jack bolts on the NW sparger.

Although analysis of the failure was incomplete, it was postulated that the failure was caused by a loss of preload due to the effect of superposition of thermal strain in excess of elastic limit due to a radial temperature gradient, upon a prestressed section of pipe under fixed bending deflection. Therefore, subsequent mechanical vibrations induced by hydraulic inputs after loss of preload, caused failure by fatigue resulting in the observed cracks in the sparger tee-boxes and wear on end pins, brackets, and thermal sleeves. However, it was noted that one of the spargers had maintained its preload, but had a minor PT crack indication.

A cold flow test was performed with the fully loaded reactor in the cold shutdown condition with the reactor pressure vessel head removed. The FW system was operated over the full operating range of flows. Two tests were performed: the 1st with the shroud head removed and without cleanup system flow; the 2nd with shroud head installed and with cleanup system flow. All 4 FW spargers were instrumented with varying combinations of accelerometers, strain gages, and dynamic pressure transducers. Also included was one dynamic pressure transducer located in each of the 4 FW lines outside the vessel.

The SE, SW, and NE spargers were removed, instrumented, and reinstalled with no change in preload. The failed NW sparger (sent to Vallecitos for metallurgical exams) was replaced with a new instrumented sparger of the same design and installed with a preload of 1700 lb. The as-tested set of spargers then represented the range of observed failures, in terms of degree of preload and cracking.

The SW sparger was observed to vibrate the most, being the only sparger to exhibit responses at discrete frequencies. Vibrations were observed in the vertical plane at 20 and 40 Hz in a mode where accelerometers in the tee-box and sparger arm were in phase. Vibration levels in the radial direction were comparable to levels in the vertical direction.

The pressure spectrum for the SW sparger arm and tee-box contained numerous sharp peaks which included structural resonances determined by shaker test. In contrast, the SE pressure spectrum contained broad-band low-level response with smaller peaks at frequencies generally above structural resonances determined by shaker test. Pressure spectra for the face plate was similar to the tee-box for the SE, NE, and NW spargers. The NE and NW indicated very little discrete frequency content although there were small blips at blade-passing frequency which were also observed on the SE and SW FW line transducers.

Preliminary analysis revealed no significant differences between signal levels in the tests with and without the shroud head/separators installed. While neither of the failures experienced posed any threat to public safety nor any detectable anomalies in plant performance, cracked spargers was an undesirable and abnormal operating condition. To gain an understanding of the problem an extensive additional program of analysis and testing was initiated to define the failure mechanixm. A 3rd set of spargers, No.3 design, was fabricated to a new design which was based on information available at the time, in order to return the plant to service while analysis efforts continued. The new spargers were instrumented with dp transducers, strain gages and RTD's and a hot flow testing program was planned.

The new spargers were of a fundamentally different support design and this was expected to reduce the possibility of fatigue failure. This was achieved by effecting a substantial reduction in operating and initial installation mean stresses by adding a primary load carrying member which provided essentially rigid attachment of the sparger to the vessel. This essentially removed the sparger as a primary load carrying member and eliminated the dependence of structural response on preloading. As a result, both the initially installed and operating mean stresses were markedly reduced (initially installed from near yield down to ~10% yield; operating stresses no longer include preload and as a result are ~25% of yield) based on then current definition of applied loads.

Applied load restraint for each sparger was provided by a bracket welded to both the RPV wall and the sparger header adjacent to the FW nozzle. This attachment provided stability to both horizontal and vertical planes. Midway between the bracket and the ends of the sparger headers, horizontal and vertical (one side only, i.e., horizontally outward and vertically upward) amplitude limiting dampers were attached to the RPV wall. Clearance between the sparger and dampers was set to a low value at assembly.

The welded attachment to the RPV eliminated the jack-bolts used for contact with the RPV in the previous design. Otherwise, with the exception of instrumentation attachment, the revised design was unchanged.

The revised design provided:

- -Solid, low-stress attachment in both vertical and horizontal planes independent of the flexibility of the sparger structure.
- -Elimination of preload requirements.
- -Increased natural frequency of the sparger structure.
- -Reduced possible high stresses resulting from high amplitude vibrations (vessel axial and radial directions).
- -Removal of the sparger as a primary load carrying member (except for normal operating loads carried by a greatly reduced moment arm fixed at the vessel).

As mentioned above the change involved the welding of brackets to the cladding of the reactor vessel interior. The bracket weldment was done after 2 layers of weld-clad were provided to build up the clad thickness to preclude the bracket weldment heat-affected-zone encroachment into base metal. Millstone estimated that it would require 75 to 100 man-rem (i.e., 30 to 40 welders each receiving up to 2.5 rem) for the welders to complete the job.

Testing and analysis of the No.3 design depended in part upon results of analysis and testing of the No.2 design directed at defining details of steady-state hydraulic input which resulted in the observed failures. Further, definitions of steady-state and transient thermal-hydraulic inputs to the Millstone spargers were required to complete the analysis. Hot flow testing data was to be recorded at prescribed intervals during normal reactor startup and during power increase to rated load. Provision was made for continuous recording of up to 15 sensor readings simultaneously for limited time periods. Major objectives of the tests to be conducted during reactor startup and initial period of operation were:

- -To measure the amplitude of possible sparger vibration under service conditions.
- -To identify possible differences in the amplitude and frequency of pressure oscillations as compared to cold test data, and identify driving functions for sparger responses in the FW system and inside the spargers.
- -To diagnose possible thermal effects which may affect sparger performance. These measurements included through-wall temperature gradients imposed on the sparger during startup and during normal operations, and the amplitude of possible water temperature variations in the vicinity of the sparger, due to uneven flow mixing.
- -To provide a comparison of thermal effects in the NW sparger and the SE sparger that are influenced by cleanup system flow in the SE sparger.

(xp)

Ref. zy offered additional information on this problem:

- It could be concluded from results of the cold tests (mentioned above) that both of the loose spargers (NE and SW) were prone to flow induced vibration of the type which involved interaction of the vibrations with the flow. There were strong indications the vibration was self-excited, rather than forced by pressure fluctuations in the feed lines.
- The 3 features (mounting bracket, midpoint bumpers and end shims) of design No.3 were expected to work individually or in combination to reduce the probability of sparger failure.
- They installed dp sensors between all 4 vertical risers that feed the spargers, specifically for operating surveillance purposes. Significant dp readings would be cause to shut down.
- Metallurgical exams of the SW and SE spargers revealed their failure mode was high cycle fatigue. (zy)

Ref. acn described the methods used to install the new spargers. Radiation measurements showed the vessel walls in the sparger area would contribute most of the dose rate during subsequent repairs. The vessel walls were hydrolasered resulting in 1 rem/hr contact with the vessel wall and 400-500 mrem/hr in the general area. At this time, the decision was made to proceed with the sparger repair work with the fully loaded core in place. An aluminum work platform was installed in the vessel to cleanup the sparger area, PT and remove the spargers. A wooden platform was constructed above the aluminum platform to prevent any subsequent tools or equipment used from falling into the core. The wooden platform was made of 14 sections of 3/4 in. plywood, "pie shaped", and held in place by a 6 in. steel ring attached to the aluminum platform. The vessel clad in the sparger area was then UT tested and cleaned with the hydrolaser. To provide for reducing the radiation level for the vessel repair work, extensive use was made of lead sheet and blankets hung on the ID of the vessel wall from the reactor vessel studs. Lead blankets were also installed on top of the wooden platform at the reactor vessel ID. Exposure was 266 R to 285 men.

The areas where clad weld build up would occur were layed out. The areas were PT tested for cracks and then ground from special clean rooms attached to the vessel wall. Welding on the vessel clad began shortly afterward. As each pad weld buildup progressed, the new weld was ground, penetrant, and ultrasonically tested.
The thermal sleeve portion of each sparger was then trued up. The 4 feedwater spargers were then placed into the nozzles for fit up of the brackets to be installed from the clad pad to the sparger. These brackets were then welded to the new pads. Final fit up of the spargers took place with shim measurements being made in the vessel. The spargers were then removed from the vessel and the end brackets welded in place. The spargers were instrumented in order to follow their operation during reactor operation. Then all 4 new spargers were installed in the vessel. (acn)

Startup occurred in mid-July after a 3 mo shutdown. The hot flow test results showed that the amplitude of sparger vibration was high at 100% power (this was [illegible] the primary cause [illegible] [illegible]. At [illegible] 80% power, vibration amplitudes were within safe limits for continued reactor operation. The [illegible] results also identified thermal cycling effects which may have been a secondary or contributing cause of some previous sparger failures. These effects were associated with the mixing of coolant flows at dissimilar temperatures. Analyses showed that the No. 3 design spargers would withstand this thermal cycling, in combination with vibration levels at < 85% power, for > 1yr without risk of failure. Due to the very shallow nature of cyclic temperature variations in the metal (only a few mills) which contributed most to the fatigue usage factor in evaluation, it was concluded that even if crack initiation were to begin, it would not propagate to a gross sparger failure without high vibratory stresses.

Vibration data from the sparger instrumentation was obtained for a period of several weeks following the completion of the planned test program. This data confirmed that the low-level vibration at 80% power was not increasing with time.

Data obtained from dynamic pressure sensors provided evidence that the dominant mode of sparger vibration was not a result of pressure fluctuations in the FW system. This evidence was consistent with that obtained from cold flow vibration tests of the No. 2 design, which were conducted during the reactor outage prior to installation of the No. 3 design spargers. It was concluded from these tests that sparger vibration was flow induced and self-excited, rather than forced by line pressure oscillations. Vibration characteristics of the No. 3 design were in some respects similar to those observed in cold flow testing, and it was believed that similar mechanisms were operative.

Efforts to better define the nature of the flow-induced vibration mechanisms were continuing, and more work was necessary to develop and confirm a solution to the problem. Power was being held at 80% and this would probably continue until the next planned refueling outage (Sept 74). A plan was developed for off-site engineering activities aimed at providing a final resolution. The plan tentatively included:

- The creation of a full scale test facility to be completed in Jan 74.
- The performance of tests in the facility on design No. 3 in Feb, on a prototype of a design No. 4 in March and April, and on a final design No. 4 in May.
- The design of thermal stress fix aspects for design No. 4.
- The development of applicable design No. 3 removal procedures.
- The design and fabrication of an instrumentation package for design No. 4.
- The design of mechanical and stress fix aspects of the design No. 4 prototype, fabrication and production of final design No. 4 replacement spargers to be completed in Aug 74. (acm,aff,afx,ans)

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N See item II. 16 for additional information.

14. CRACKS IN STE[illegible]

Nine Mile Pt. 1 - Ma. 73 & [illegible]ring 74

Cracks were discovered in the steam dryer assembly. The assembly is mounted in the reactor vessel above the steam separator assembly. It forms the top and sides of the wet steam plenum and provides a seal between the wet steam plenum and the dry steam flowing out the top and down the steam nozzles. Moisture is removed by impinging on the dryer vanes, flowing down through collecting troughs and drain tubes to the reactor water in the downcomer annulus below the steam separators.

The initial discovery resulted from a visual examination which revealed that a separation had occurred in the top longitudinal weld joining the No. 3 diffuser to its vertical support plate. This crack extended along the weld centerline for ~ 24 in. then curved downward about 3 in. in the vertical support plate. GE was requested to investigate the problem and coordinate repair procedure.

The steam dryer was removed from the reactor pressure vessel and positioned underwater in the st[illegible]ge area. Support structures were erected. GE's examination of the dryer structure employing underwater TV scanning revealed a total of 8 cracks varying in length from 1 in. to 2 ft in 3 steam diffuser weldments. Extent of the observed fractures were limited to the top longitudinal weld seams except for the 3 in. extension of the one 24 in. crack into the vertical support plate. The weldment design consisted of 1/2 in. top plate joined to 1/4 in. vertical plate by full strength corner welds.

From a review of the video tape, particularly high resolution views certain observations of weld conditions and crack topography could be made. Both the central divider plate and relatively thin diffuser support plates at weld seams had sustained buckling apparently from constraint preload effected by the stiffer top plate and outer shell structure. The majority of cracking developed in those weld seams where excessive surface grinding was performed. Although, it was not possible to locate initiation points, the cracking tended to follow either the weld centerline or fusion line in the vicinity of buckled plate areas. In the case of complete separation of the weld joint, the crack exhibited brittle fracture features of extreme linear nature and was not accompanied by any discernable secondary cracks or pitting typical of corrosion attack.

There was no evidence from all recorded data which would implicate water quality as a causive factor of the failures. GE postulated that poor workmanship during steam dryer fabrication and the reaction stresses from effects of constraint preload and operational vibration probably caused weldment failure by fatigue.

The repairs were to include excavation and rewelding of ruptured areas and contour-blending for conformance to the structural design specs. In addition, sized canning plates (angle type) were to be installed at equal intervals along affected weldments of all 8 diffusers to reinforce the structure and reduce the possibility of fatigue failure under service [illegible]ditions. PT acceptance inspection was to be exercised throughout the repair operations.

Plant management considered that the dryer imposed no significant safety risks, however, the dryer assembly was to be reinspected during the next scheduled refueling outage to verify the adequacy of the modifications. (xb,agw)

During the Spring 74 refueling outage they found additional cracks in steam dryer chambers. They were caused by arc strikes and insufficient welds. The affected areas were replaced. (ayh)

15. INADEQUATE STEAM SEPARATORS & DRYERS

Würgassen - 1971-72

The cyclone separators are arranged in an annular configuration so they can be left in place during refueling operations. The chevron dryers are moved for refueling. Commissioning tests [illegible] to confirm the results of the developm[illegible] At 65% power the steam moisture con[illegible] limits. They attributed this t[illegible] separation in the cyclone's [illegible] a problem with the design o[illegible] dryer inlet area which result[illegible] dryers, and 3) inadequate dime[illegible] discharge area. (a[illegible]

[illegible] BWR-2 II. Reactor Int. P. 8.

Ref. zy offere[illegible] problem:

-It could [illegible] tests [illegible] sparge[illegible] vibr[illegible] of [illegible]

The moisture content of the live [illegible] rapidly with increasing power. Ef[illegible] reduce the moisture content by modi[illegible] the steam dryer assembly and the stea[illegible] failed. Power was then limited to ~ 6[illegible] New dryer assemblies and steam separato[illegible] ordered. They were installed in 1974 wh[illegible] plant was shutdown because of turbine pro[illegible] (awl,brw,bus)

16. FW SPARGER CRACKS

Millstone 1 - 1972, 73

See items II.11 and 13 for initial information. [illegible] conducted a study of FW spargers to determine wh[illegible] cracks appeared in the spargers at Millstone. T[illegible] investigation included inspections of spargers a[illegible] operating plants, instrumentation of the Millsto[illegible] spargers and cold flow testing at a test facilit[illegible] Inspection of spargers in 10 operating plants wa[illegible] completed. The only plant with confirmed defect[illegible] was at Millstone. Tests performed at the test facility showed there was a relationship betwee[illegible] thermal sleeve/nozzle leakage and sparger vibra[illegible] For the cold flow sparger tests, the vessel noz[illegible] ID at the thermal sleeve fit was increased in i[illegible]crements for the performance of leakage tests a[illegible] corresponding full flow tests. These tests yie[illegible] the following: 1) For a given sparger flow, th[illegible] leakage was directly proportional to the avera[illegible] radial gap between the nozzle and thermal slee[illegible] and 2) For a flow of 4500 gpm, the sparger was susceptible to self-excitation or leakage flow induced instability if the average radial gap > .011 in. These results were obtained by tes[illegible] with no preload on the sparger and the retaining pins at the ends of the sparger arms were essentially loose except for the reacting force against the hydraulic flow. The only variable introduced was the change in the average radial gap between the nozzle and thermal sleeve (i.e. leakage area varied).

Based on this, GE recommended the following: 1) Operating Plants - Perform inservice inspections during refueling outages to determine the conditions of spargers; 2) Near Operating Plants - By expanding the sparger thermal sleeve, the radial gap between the nozzle and thermal sleeve could be reduced to a point where instability could not occur. The sparger should be removed and expanded, as necessary, to give the proper radial gap to eliminate flow induced vibrations; and 3) Future Plants - For those plants where vessel and thermal sleeve fabrication and installation schedules allow, the thermal sleeve was to be welded to the nozzle safe end, the thermal sleeve design was similar to the advanced welded design for use on BWR/6, no change was required. (akb)

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17. EXCESSIVE FW SPARGER CLEARANCE

Cooper - Jan 74 (prior to initial fuel loading)

As part of the GE evaluation of cracking in FW spargers at Millstone (see II.13), it was discovered that there was excessive clearance (about 11 mils) between the thermal sleeve (which is a single pipe attached to the FW sparger) and the reactor vessel nozzle into which the sparger thermal sleeve fits. When the radial clearance exceeds about 6 mils a vibration occurs that contributes to cracking of the sparger and nozzle wall. GE developed a fix which consisted of 1) removing the spargers from the nozzles (done from inside the vessel), 2) measuring the diametral gap between thermal sleeve and nozzle, 3) expanding the thermal sleeve to make a tighter fit, and 4) re-insertion of the thermal sleeve into the nozzle. The sparger and interior of the nozzle could be inspected for cracking during the procedure. GE was going to recommend this be done on all plants of this design, including operating plants.

GE recommended this procedure be accomplished prior to startup of Cooper. They agreed and deferred preparation for fuel loading. The 4 spargers were removed and found to have diametral gaps of 6 to 13 mils. The pipes were expanded to a negative clearance of 1 to 4 mils to assure tight fit and reinstalled. The operation took < 1 wk. (ajm)

18. FAILURE OF BLADE GUIDE YOKE

Arnold - Feb 74 (prior to initial fuel loading)

During operator training, action was undertaken to lift 2 blade guides from a storage location in the dry fuel pool. When the 2 blade guides, connected by a yoke at the top, had been lifted about 3 in., the yoke handle broke. The blade guides fell back into the storage location and a piece of the handle fell to the floor of the fuel pool. All pieces were retrieved. The yoke is cast aluminum.

PT of the handles of the blade guides connected by the failed yoke indicated a crack in one corner of one handle.

All remaining blade guides were removed for individual tests and exams. All blade guide handles and yoke handles were loaded to 1.5 times static load and then PT examined for indications of cracks. One blade guide handle and one yoke handle were rejected due to indications of porosity. Only guides and yokes with tested and accepted handles were returned to the vessel. (ano)

19. JET PUMP FAILURES

Quad-Cities 1 - Apr 74

A scheduled jet pump inspection was performed during refueling operations. This was carried out with the aid of underwater TV and a video tape unit. The inspection included:

- Inspection of the tack welds on the beam bolt keepers and a check of beam pocket fits.
- Observation of the beam bolt assembly while applying 200 ft-lb of torque to the beam bolt in both directions.
- Inspection of the restrainer gate and wedge positions and restrainer bolt tack welds.

The following discrepancies were noted:

- While applying up to 200 ft-lb to the beam bolts, movement was detected on 3 bolt assemblies, (10, 13, and 16). They failed the test.
- 3 restrainer gate bolts and keepers were found to be completely sheared and missing, 2 of these were from pump 5; one was from pump 6.
- 30 of 37 remaining restrainer bolt keeper welds were either missing or cracked.
- The restrainer gate wedge on jet pump 5 was missing.
- The restrainer gate on pump 18 showed signs of wear in the vicinity of the wedge indicating possible vertical movement of the wedge prior to the inspection.

A subsequent inspection of the inlet riser braces for pumps 5 and 6 showed the brace to vessel welds to be intact.

The following parts were missing: 3 restrainer gate bolt heads, 3 restrainer gate bolt head keepers, 1 wedge, and 1 wedge spring. Attempts to locate and retrieve the missing parts retrieved the following: 1 restrainer gate bolt head, 2 restrainer gate bolt head keepers; 1 wedge in 3 pieces (wedge, wedge blade, and wedge spring); and 1 additional wedge bale.

An extensive plan was developed by the station and GE to repair the discrepancies. All 20 beam bolts were to be completely detensioned and then retensioned. The procedure was one previously used at Quad-Cities for jet pump repairs but modified from the tensioning procedure used for the original installation. It included observing the beam bolt for rotation while being tensioned. The retensioned beam bolts were then to be rewelded with 2 tacks per bolt.

All 20 restrainer gate assemblies were to be replaced with new assemblies. These were to be installed, properly tensioned and restrained, and the keepers welded with one tack per bolt (total of 40 bolts).

The riser braces on all inlet risers (1 riser/2 jet pumps) were to be completely inspected by using underwater TV. This primarily checked the brace to vessel wall welds for integrity.

Pumps 5 and 6 were to be removed and raised to a position for more complete viewing. A complete inspection was to be performed.

Attempts were to be made to locate and retrieve the missing parts. This included 2 restrainer gate bolt heads and one restrainer gate bolt head keeper. Additionally, an attempt was to be made to determine the source of the extra wedge bale.

All parts retrieved from the vessel were sent to GE for exams to determine the possible causes or reasons for the failures.

Even though there were significant failures of restraints and fasteners, there were no jet pump failures or loss of integrity. Particularly there were no failures of any jet pump holddown components. Thus there were no unsafe conditions existing during previous periods of operation.

The cause of the failed restrainer gate bolt keeper tack welds and the apparent lack of preload on the beam bolts was attributed to construction and installation deficiencies. The apparent cause of the failed restrainer bolts and the missing and possibly moved restrainer gate wedge was probably excessive vibration as a result of the deficient installation. The problems discovered were similar to those discovered during inspections related to the failure described in II.8. It appeared that all jet pump problems experienced at Quad-Cities could probably be traced to faulty craft installation and workmanship during initial installation.

A previous failure of flow instrumentation for pump 7 on Unit 1 appeared to have a reasonable explanation and could be related to the failures. That failure was located as a severed instrument line at the point where the line was attached for support to pump 6. It appeared the failure could have been caused by excessive vibration of pump 6 due to the failed restraints. (aox)

Ref. aqr provided additional information. The restrainer gate wedge was missing on pump 6 (not 5). Several of the original restrainer gate wedges fit in place too deeply and did not meet the installation tolerance specified. The wedges for the new gate assemblies were redesigned to be slightly larger in size. The larger wedges were expected to assure the required tolerance would be met.

The restrainer gate bolt keepers were also slightly redesigned. The new design eliminated a shoulder and lip on the edges of the keeper. Those edges caused interference and damage to the tip of the welding electrodes used to tack the keeper in place. The lip served no useful purpose and was eliminated to aid in the welding operation.

Riser braces on all 10 jet pump inlet risers were inspected with underwater TV. All brace to vessel wall welds were intact.

Jet pump 5 was lifted ~ 2 in. from its normal position. The seating surfaces at both the inlet and outlet were inspected with TV and were found in normal condition. Pump 6 was lifted from its normal position and raised completely out of the water for inspection. Both the inlet and outlet seating surfaces were normal. Slight wear on the order of a few .001 in. was observed on the belly band in the vicinity of the contact area with the wedge. (aqr)

N Ref. asy provided additional information. Several design changes were made to the replacement parts to facilitate remote repairs while restoring the pump assemblies to their original functional requirements. The basic design remained unchanged. The design changes were incorporated primarily in assembly features and techniques.

N Restrainer gate assemblies for all 20 pumps were replaced with new assemblies. The entire assemblies were replaced rather than just the failed bolts in order to minimize radiation exposure to personnel. The restrainer gate wedges for the new assemblies were slightly thicker to assure the installation tolerance was maintained. All newly installed wedges met the specified installation tolerance. Redesigned restrainer gate bolt keepers were installed on all 40 restrainer gate bolts. Two tack welds per gate bolt keeper were applied instead of the one weld specified for the original assembly, to provide a greater margin of integrity.

N A standard restrainer gate required special modification in order to facilitate installation on Pump 5. The standard gate was cut in 2 at a point 45° from the center line of the wedge on the side nearer the hinge pin. A spacer block was then welded between the 2 parts in order to provide an offset of 1-3/8 in. The offset was necessary to compensate for one side of the jet pump hold down bracket which was bent. Because of the very close proximity of the hinge pin to the inlet mixer of the jet pump, a new design bolt and keeper were provided in order to avoid interference of the bolt and keeper with the mixer.

N A standard restrainer gate was also modified for installation on Pump 6. An extension arm was added to the gate on the hinge pin side. The extension arm had a locator screw assembled in it to replace the locator block and adjusting screw that were broken off. The extension was assembled to a new design gate bolt and the existing hinge pin, and was secured by a new design nut. The nut was prevented from rotating by a keeper which was tack welded to the extension arm. The bolt was kept from rotating via a Right-Hand and Left-Hand thread design feature. The locator screw was adjusted to mate with the belly band on the inlet mixer assembly and tack welded in position.

N The 2nd locator screw on Pump 6 had been found with a broken tack weld and was not properly positioned in its locator block. This screw was returned to its proper location as indicated by the original tack weld and a new tack weld was applied. Pump 7 was found with one locator screw missing from its mounting block. A new locator screw was designed to facilitate remote assembly and installed, adjusted and tack welded in position.

N Beam bolt retainers were found missing from the beam bolt assemblies of Pumps 10, 15 and 16. These retainers were not replaced since their only function is to prevent the beam bolt assembly from falling from the inlet-mixer during assembly or disassembly. Any future disassembly of these inlet mixers will require the beam bolt assemblies be removed as separate assemblies. The beam bolts for all 20 jet pumps were retensioned using the procedure developed during the jet pump repairs to Unit 2 in 1972. This required a minimum bolt rotation at a torque value of 50 ft-lb which provided assurance the specified tension was applied. Following re-tensioning of the beam bolts, the beam bolt keepers were secured with 4 tack welds/keeper instead of the original 2 tack welds/keeper to provide a greater margin of integrity.

N The failure of the flow instrument line for Pump 7 was identified as a severed line during this outage. Because of the engineering and mock-up required to develop the necessary tools to repair the instrument line, the fix for the severed line was not available for that outage. It was planned to operate during the subsequent cycle with the instrument line as is now, and to try to repair it during their next refueling outage.

N The following loose parts remained in the vessel: 1 Retainer for beam bolt ("L" Shaped piece 1/8 in. thick, 2 in. wide, one section 2 in. long, one section 3 in. long), 2 Bolts for above retainer (3/4 in. long x 1/2 in. dia), small piece masking tape (4 x 2 in.), LPRM spring reel guide piece (1 in. dia x 2 in. long), short piece of welding rod (6 in. long), sleeve from refueling shield-cylinder (3/4 in. dia gate (cattle chute), 1/2 in. long). The probability of flow blockage from the pieces was considered small.

N The conclusion of the analysis of the inspections and repairs was the failures were fatigue failures caused by excessive vibration. The vibration was

... several of the jet pump [illegible] not been tensioned to [illegible]et the [illegible] requirements. Finally, the failure [illegible] the original design requirements was [illegible] improper field installation of th[illegible] (asy[illegible] involved.

Millstone 1 - Sept 74 - refueling shutdown

See items II. 11, 13 & 16 for details on previous information. Two sets of FW spargers had failed in service. The failure of the original Design 1 sparger set was found during the inspection which followed the chloride intrusion incident of Sept 72. At that time it was concluded that failure was due to high cycle fatigue as a result of inadequate FW sparger preload. The replacement spargers, Design 2, were essentially of the original design but were installed with a much higher cold preload. After the replacements were operated ~ 6 wk, inspections again showed fatigue failures.

The 3rd set of spargers, Design 3, was installed in June 73 and was in service until late Aug 74. These spargers featured a fundamentally different support bracket attached to the vessel wall near the FW nozzle. Instrumentation installed on the spargers showed excessive vibration when the plant was operated at > 85% power. Power was subsequently limited to 80% to avoid excessive sparger vibration.

During the scheduled refueling outage which began in late Aug 74, an inspection of the spargers revealed that the 2 uninstrumented spargers and associated support brackets were badly cracked. PT of the bracket cracks was extended to include the cladding of the FW nozzle corner radii due to the proximity of the support brackets. Several linear indications were noted in the nozzle area. Therefore, PT of the entire clad area adjacent to all FW nozzles was performed revealing a total of 23 indications. Concurrent with these exams a clad defect removal procedure was prepared.

In conjunction with the removal of the clad defects, a metallurgical boat sample was extracted from one of the cracks for analysis by GE. In addition, Teledyne performed an analytical evaluation of the cracks and ITT Grinnell reviewed the NDT crack removal and crack measurement procedures and data for adequacy.

The majority of the cracks were ~ 1/8 in. in depth, the largest being ~ 7/16 in. deep and 2 in. long on the NE FW nozzle. The majority of the clad cracks were oriented radially outward from the longitudinal axis of the FW nozzle on the reactor vessel shell to nozzle corner radius.

All cracks were mechanically removed by grinding in small increments followed by PT. The crack lengths diminished gradually with the depth of grinding, typical of a fatigue type crack. Crack depths were measured by spanning the crack surface with a straight edge from which depth micrometer readings were taken. Additionally, wax impressions of 2 cracks were taken from which depth measurements as well as crack profiles could be determined. The final ground areas were blended to a 3 to 1 taper to eliminate any surface discontinuity, and a final PT was performed to ascertain that the cracks had been completely removed. In conjunction with the grinding restrictions used to assure that the low-alloy steel base metal was not touched, one of the deeper ground areas was etched revealing only SS clad metal.

All 4 FW nozzles were UT examined from the exterior of the reactor vessel during the outage and no reportable indications were detected.

The exams and analyses indicated the cracking in the vessel shell to nozzle was limited to the SS cladding and resulted from high cycle thermal fatigue attributable to excessive bypass flow around the previous FW sparger thermal sleeves. It was believed this bypass FW flow caused rapid temperature fluctuations in the affected area, a phenomena which was expected to be minimized by the interference fit of FW sparger Design 4 (installed in Fall 74). The interference fit was based upon measurements of the dia of the thermal sleeves and the nozzle bores. This procedure was expected to assure continuous contact around the circumference of the thermal sleeve, except during intermittent periods of very low FW temperature (100°F) experienced during startup Exams prior to assembly did not reveal points of severe local discontinuities. This operating condition, associated with reduced FW flow, produces an average radial gap of ≤ 0.003 in. As a result of these design improvements virtually no leakage through the gap was expected during full power operation. The anticipated leakage at low power levels was not expected to cause significant annular fluid temperature fluctuations in the thicker regions of the nozzles, because the small amount of leakage flow would be heated as it passed through the annulus.

A fracture mechanics analysis based upon ASME Section XI was performed to determine the permissible flaw depth and in no case did the detected cracks equal or exceed the permissible flaw.

In order to understand the nature and cause of the original FW sparger vibration problem, a full scale test facility was built in San Jose. A test program was conducted which aided in isolating the cause of vibration. Design 4, evolved from the information obtained from the test facility. A set of Design 4 spargers was installed in Millstone during the Fall 74 outage. Following a brief period of confirmatory test data acquisition during plant startup, the unit was to be returned to full power operation.

The cold flow testing in the full scale test facility identified the cause of sparger vibration to be excessive clearance between the sparger thermal sleeve and the vessel safe-end. For a given pressure drop across the sparger there is a critical flow area (or radial clearance for a given size of thermal sleeve) which, if exceeded, permits sparger vibrations. The results of the test program indicated the failures of Designs 1 and 2 and the excessive vibration of Design 3 were caused by excessive clearance between the thermal sleeve and the vessel safe-end.

The Design 4 interference fit concept was proven by cold flow testing in the test facility, and was expected to perform satisfactorily. Verification of this design was to be accomplished during initial plant startup using data from an instrumentation package installed on the spargers. Based on the test results, the salient design features of the Design 4 spargers were:

1) Interference fit between the sparger thermal sleeve and vessel safe end: This was based on several tests showing there was no vibration under any flow conditions when the thermal sleeve was tightly fitted to the safe end.
2) Forged-welded tee between the thermal sleeve and sparger headers: This reduces peak stress levels

in the tee by a factor of 4 due to smaller stress concentrations. This is due to the use of full penetration welds, more uniform sections, and large radii at the junction of the header pipes and the thermal sleeve.

3) Larger flow distribution exit area: This was incorporated to lower the pressure drop at rated flow from 17 psi for the first 3 designs to 10 psi for Design 4, which increases the stability margin.
4) Sparger installed preload of ~ 2600 lb: This was incorporated to inhibit sparger radial motion due to FW flow. This was accomplished by installing the sparger with a preload that exceeds max expected FW flow forces.

Each Design 4 sparger was provided with vibration instrumentation. Only one sparger (SW) was to be fully instrumented with numerous strain gauges and RTD's. Due to the importance of detecting local temperature fluctuations in the region of the nozzle to vessel shell radius, RTD's were installed on all 4 FW nozzles prior to plant start-up. Data taken from those sensors was to be monitored and recorded in a manner similar to the vibration monitoring planned for the hot flow test program which would correspond with the plant's ascent to power following the outage.

UT of all 4 FW nozzle corners from the reactor vessel exterior were planned after ~ 6 mo of operation. All indications detected at that time were to be satisfactorily dispositioned prior to return to power. During their next refueling outage, UT and PT was planned on the accessible nozzle corner cladding of all 4 FW nozzles concurrent with an exam of the spargers. UT, identical to that performed at the 6 mo interval was also to be conducted during that outage.
(axp)

Ref (bbr) provided additional information. Originally they concluded that they had 23 cracks, none of which penetrated into the base metal. A revaluation of the deeply ground areas was planned and conducted subsequent to the fuel reload operations. This was necessary because all deeply ground areas had not been chemically etched and the apparent clad thickness was substantially in excess of that specified in the original design documents. Following vessel flooding and subsequent to the fuel reload operations, the water within the reactor vessel cavity was drained to a level just below the FW spargers and the vessel work platform was installed. A visual exam of the previously ground areas adjacent to the 4 FW nozzles revealed 3 areas of rust (ferric oxide). The 3 areas were on the nozzle radii of the SE, NW and NE nozzles. This represented exposure of the low-alloy steel base metal by the crack removal operations.

The reference also stated that the correct number of detected cracks was 20.

Subsequent to the visual exams, PT was performed on the NE FW nozzle grind area, UT of clad thickness was performed at ~ 35 locations adjacent to the deeply ground areas, 2 replicas of each rusted area were made and depth measurements of the base metal penetration were performed. In addition, Teledyne performed an analytical evaluation of this base metal penetration and ITT Grinnell examined the rusted areas and advised with respect to replica formation procedures.

A review of the Reactor Pressure Vessel Stress Report was conducted, the conclusion being that the amount of base metal removed did not detract from the original design requirements. A fracture mechanics analysis indicated the calculated critical flaw depth was not approached. Removal of the cracks assured that discontinuities, capable of propagation, did not exist at that time. The cause of the cracking, the planned surveillance and the corrective action associated with FW sparger Design 4 remained the same. Implementation of those measures was expected to assure a safe return to full power operation.
(bbr)

R 21. FW SPARGER CRACKS

R Muhleberg - Aug 73 and summer 1974 - refueling shutdown

They inspected their spargers after the discovery of cracks at Millstone. Some indications of very small cracks were found at that time but continuation of operation was still considered practical. A close visual inspection with the aid of a crane operated chair showed larger cracks in 2 of the 4 spargers in the summer of 1974. In a close collaboration between GETSCO and BKW the replacement program was very carefully prepared. It involved among many things the fabrication of a shielded platform and shielding curtains for the vessel walls in order to reduce the strong radiation (1 R/hr) in the working area.

The 4 FW spargers were fabricated before the outage (they had to be modified slightly). The whole replacement work was performed by a team of 27 specialists of Mannesmann and BKW (the owner) and GETSCO supervisory personnel.

In Sept and Oct 74, a preparation program for replacing the spargers was carried out by GETSCO and Muhleberg personnel. It involved planning the necessary operations, reworking the previously ordered new spargers, fabrication of a shielded platform and shield curtains for the vessel walls and employment of some 25 skilled men.

Shutting down the plant, opening the vessel and installing the platform and shielding was completed within 2 1/2 days. Work started in Oct with the removal of the old spargers. One of them had a circumferential crack. Indications of slight cracking were also observed on the 2 spargers which were still sound in the summer inspection.

The installation of the new spargers involved a careful adjustment of the thermal sleeves to the dia of the FW nozzles. The gap between nozzle and sleeve was determined as a critical parameter for the possibility of vibrations. Detectors for sound emission were installed at the outside of the nozzles.

As planned the whole operation was finished after 6 days. The accumulated exposure of all people involved was 44 man-rem. The duration of the

R shutdown was ~ 1 1/2 wk. (bah,bfo,bqg,btk)

22. VIBRATION - FW SPARGER U-BOLT FAILURES

Humboldt Bay 3 - Nov 74 - shutdown

The reactor FW sparger consists of a continuous section of oval cross section pipe located around the inside of the vessel at an elevation below the top of the core chimney. The sparger is supplied with water from one inlet nozzle. It has 1/2 in. holes near the bottom on both sides to distribute the FW uniformly in the vessel. The sparger is supported by 8 gusset type brackets around the ID of the vessel wall and was originally designed to be held in place with U-bolts mounted on each bracket. The U-bolts, which were fabricated from 1/2 in. diameter SA-276 Type 304 austenetic SS, fitted around the sparger and were held in place with nuts on both sides of the support bracket. The nuts were tack welded to the bracket.

During a refueling outage, it was noted that the sparger had moved 1 1/2 to 2 in. away from the nozzle and some of the U-bolts were broken. A close exam showed that 6 of the U-bolts had broken through on one or both legs. Two bolts were sent to GE for metallographic exams. These bolts were broken through on both legs just above the nut on the top of the support bracket.

The sparger thermal sleeve still extended ~ 22.6 in. into the nozzle. The sleeve had an interference fit and the nozzle ID is uniform so there would not have been an increase in FW flow around the sleeve.

GE's exam procedures consisted of visual, PT, scanning electron microscopy, metallography and hardness. The exams showed the failures were caused by high cycle fatigue. It was postulated that the U-bolts vibrated in resonance with input from a source such as the FW pump or recirc flow. It was further postulated that the sparger itself was not vibrating or that the vibration amplitude was very low.

Corrective repairs were performed under water using specially designed tools. All of the U-bolts and associated nuts were removed from the support brackets. There were no lost or unaccounted for parts. Basic tools used were a commercial hydraulic bolt cutter and nut cracker modified for the application. The sparger was then recentered using a commercial hydraulic jack with special fixtures built at the plant so the sparger could be jacked against 2 of the sparger support brackets. The force required to move the sparger back in place was calculated to be < 2000 lb.

After the sparger was recentered, the FW line inside of the drywell was examined by PT. No relevant indications were found. The FW nozzle was also examined by radiography. This showed that the thermal sleeve which attaches to the sparger was in its proper position. The end of the thermal sleeve appeared to be normal in all respects.

Redesigned sparger restraints were installed at each support bracket. These restraints were substantially stronger in design and clamped tightly around the sparger. They were not directly attached to the sparger support brackets as were the original U-bolts. The nuts which bolt the restraints to the sparger were tightened to 40 ft lb and secured by staking to prevent them from working loose. Legs from the restraints were designed to engage the wall bracket to limit any sparger movement away from the nozzle due to hydraulic forces, while allowing the sparger to move as a result of thermal growth.

The new restraint design was not expected to be susceptible to the same types of vibration problems postulated for the original U-bolt design because the new restraints were held in firm contact with the sparger. The new restraints were fabricated in the plant from solution annealed ASTM A-276 Type 304 SS bar and flat stock.

Twenty-three days elapsed from the time the sparger problem was first discovered until the corrective repairs were completed. All of the repair work was performed by plant personnel working from the refueling platform on the reactor extension tank to minimize radiation exposure. The total accumulated exposure was ~ 6.5 man-rem. All special tools and the new restraints were built by plant personnel. Engineering support was provided by GE.

During the 1975 and 1976 refueling and maintenance outages a thorough visual inspection of the FW sparger, all support brackets and all sparger restraints was to be made. In addition, each restraint was to be checked to assure it was still clamped tightly to the sparger. During the 76 outage one of the restraints was to be removed and replaced by a new restraint of similar design. The removed restraint was to be subjected to a thorough exam. (bes)

23. FW SPARGER CRACKS

Dresden 2 - Dec 74 - refueling shutdown

FW sparger (304 SS, Sched 40) inspection was performed in response to a GE recommendation. The inspection consisted of using underwater TV to examine areas including: 1) welds of 6 in. schedule 40 header pipes to the 11 in. junction box pipe, 2) weld of junction box to thermal sleeve, 3) contact of bearing bars to vessel wall, 4) pin engagement with clevis ends of each sparger and 5) clad FW nozzle blend radii area on the reactor vessel. The inspection revealed 2 cracks. The first crack was located on the upper part of the right side header pipe to junction box weld area of the south-west quadrant sparger. The crack appeared to be relatively straight, extending 90° around the pipe circumference. The 2nd crack was located on the upper part of the left side header pipe to junction box weld area of the north-east quadrant header. This crack appeared to be a little more jagged than the first crack, near the weld area and extended ~ 200° around the pipe circumference.

The bearing bars (preload spacer), on the right side of the south-east quadrant sparger and on the right side of the north-west quadrant sparger were not in contact with the vessel wall. No other discrepancies were observed.

Based on the underwater TV inspection, the predominate failure mode appeared to be fatigue cracking. The video tapes of the sparger inspection were viewed by GE personnel and concensus of opinion was the cracks looked very similar to the sparger cracks at Millstone 1 (see II.20). The cracked spargers at Millstone were metallurgically examined and determined to be transgranular (fatigue cracks).

As viewed through TV, the bearing bar on each of 2 spargers appeared to be out of contact with the vessel wall by a slight amount. Although there were several possible explanations for this, the actual cause was not known. This condition had been observed at other BWR's and its presence did not alter the conclusion that the sparger cracks were caused by fatigue due to flow induced vibrations.

Numerous full-scale cold-flow test had been conducted at a GE test facility on FW spargers of several different configurations to determine the cause of vibration. Those tests showed that unstable flow-induced vibration occurred as a function of the following variables:

- The dp between the sparger inlet and discharge.

- The average radial gap, which permits leakage flow between the inside of the thermal sleeve and the outside of the FW nozzle.

- The amount of damping present in the system, particularly at the thermal sleeve-to-nozzle interface.

The program for corrective action was to 1) PT the accessible portion of the nozzle blend radius of each FW nozzle, 2) further inspect the old Dresden 2 spargers and determine the actual radial gap when they were removed, 3) replace all 4 Dresden 2 spargers that outage with new spargers of the new design (very similar to the [illegible] spargers at Millstone), 4) inspect the FW spargers during their next scheduled refueling outage, 5) inspect (including PT of the nozzle blend radii) the Dresden 3 FW spargers during its next scheduled refueling outage.

The salient design features of the Design 4 FW sparger were:

1) Interference fit between the sparger thermal sleeve and vessel safe end.

2) Forged-welded tee between the thermal sleeve and the sparger headers.

3) Different size and location of exit holes in the sparger.

4) Schedule 80 rather than schedule 40 (304 SS) spargers. (bfn)

N There were ~ 200 men used to replace the spargers. (cvc)

24. [illegible]

[illegible] - Feb 75 - refueling [illegible]

A scheduled jet pump inspection was performed with the aid of underwater TV and video tape. Upon completion of inspections on all 20 jet pumps, the following discrepancies were noted: 1) The beam bolt retainer clips and 1/2 in. cap screws were found to be missing on jet pumps 7 and 8; 2) The restrainer adjusting screw on the shroud side of jet pump 1[illegible] was missing; and 3) Restrainer gate keepers on jet pumps 1, 8, 12, 15, and 18 had their welds in place, but not fused to the restrainer gate.

The apparent cause of the missing retainer clips, the 1/2 in. cap screws, and the adjusting screw was attributed to equipment failures caused by installation deficiencies. Based on the parts being inadequately installed, the vibrational forces present during normal operation could have caused those components to become dislodged from their normal positions.

During jet pump repairs to Unit 2 in 1972 (see II.8), the beam bolts and keepers were the primary components repaired. The repairs necessary to Unit 1 during its past refueling outage in 1974 revealed that further defective installations (see II.19) existed. It was felt that their then present components were originally installed in a deficient manner and the possibility of their failure was not known at the time of the original Unit 2 repairs. The discovery of their new failures was consistent with the deficiencies noted during the Unit 1 outage in 1974.

The missing beam bolt retainer clips and 1/2 in. cap screws of pumps 7 and 8 were not replaced. Those pieces were important only during the initial construction of the jet pump beam assembly or during their removal. After installation of the beam bolt, the function of those pieces was completed.

A replacement stop (adjusting) screw was installed and tack welded to its holding clamp on pump 18. As an alternative to removal of the jet pump throat, the welding tool used to tack weld the stop screw was modified to perform its intended function.

Although it was postulated that the restrainer gate keeper on pump 18 was loosened during the repair of the stop screw, the tack welds of all jet pump restrainer gates were reinspected as a precautionary measure. As a result of the faulty tack welds discovered on the jet pump restrainer gate keepers of pumps 1, 5, 12, 15, and 18, the restrainer gate bolts were retorqued, and new tack welds were placed and verified. A similar inspection of the 20 jet pump beam bolt keepers was performed with satisfactory results.

As of late Feb 75 there were 11 objects (dosimeter, screws, clips, wire, wrench, rag) loose or missing in the reactor vessel. Search and retrieval attempts were continuing.

The problems discovered on this inspection of Unit 2 were similar but less severe than those discovered following a jet pump failure in Unit 2 in Aug 72. At that time, a jet pump assembly had become dislodged from its normal position and rotated in the vessel. Extensive inspections and repairs were performed on all Unit 2 jet pumps and they operated satisfactorily thereafter.

An extensive inspection of all Unit 1 jet pumps during its first refueling outage in Apr 74, revealed a large number of jet pump discrepancies. These included beam bolt torque test failures, sheared restrainer gate bolts and keepers, missing and cracked restrainer gate bolt keeper tack welds, a missing restrainer gate wedge and indications of wear in the vicinity of the wedge indicating possible vertical movement of the wedge prior to the inspections.

All of the jet pump problems which had occurred at Quad-Cities were attributed to faulty craft installation and workmanship during the initial construction of both units. (bhp)

At the end of the outage a neutron dosimeter tube, a neutron dosimeter, an adjusting stop screw and a grinding burr were still missing and assumed as remaining in the vessel. (bui)

25. FW SPARGER CRACKS

Quad-Cities 2 - Feb 75 - refueling shutdown

PT of the 4 FW Sparger Junction Boxes revealed indications of cracking. The test was being performed upon request of GE because of concern over the occurrence of cracking in several other FW Spargers of the design utilized in Quad-Cities. Special attention had been given to the welds at the intersections of the junction boxes and the distribution pipes. The PT tests revealed indications of several cracks in the heat affected zones on the piping side of the welds on each of the 4 spargers. Following removal of the old spargers, further PT exams were conducted on the FW nozzle cladding with the following results: 1) On the 60° nozzle, 3 porosity indications were found, 2) On the 150° nozzle, one fine crack, 3 linear indications, 1 crater and scattered porosity indications were found, 3) On the 240° nozzle, one linear indication was found, and 4) On the 330° nozzle, 8 linear indications were found. All relevant indications were removed by grinding and retested satisfactorily.

The failures were attributed to fatigue to the spargers caused by flow induced vibration, and compounded by stresses induced by the thermal gradients inherent between the FW piping and reactor vessel internals. There have been 36 startup/shutdown cycles. Leakage between the sparger and the FW nozzle contributed significantly to vibration of the sparger assembly and also imposed thermal stresses on the nozzle.

The failures were similar to those at Millstone and Dresden 2 (see II.23, 32 and 33). Spargers of a new design were installed in Quad-Cities 2. They utilized an interference fit to eliminate leakage and thus reduce the vibration. The old spargers were to be further inspected to determine the actual radial gap that existed during operation. The spargers for Unit 1 were to be inspected during its next scheduled refueling outage. The newly installed spargers for Unit 2 were to be inspected during its next scheduled refueling outage in Sept 76. (bho,cmx)

26. INSTRUMENT VIBRATION - CHANNEL BOX DAMAGE

Fukushima 2 - Spring 75
Cooper - Apr 75
Muhleberg - 1975
Pilgrim - 1975
Brunswick 2 - 1975
FitzPatrick - 1975
Arnold - 1975
Peach Bottom 2 & 3 - 1975
Hatch 1 - 1975
R Browns Ferry 1 & 2 - 1975
Vermont Yankee - 1975

GE reported that TIP systems had provided indications of possible vibrations at Fukushima 2 and at Cooper. Preliminary exams by GE indicated that the coolant flow through bypass holes in the plate supporting the reactor core could be inducing movement of the instrument components sufficient to contact the corners of the channel boxes, causing wear and cracking. Such wear and cracking were observed in Fukushima 2. Channel boxes are structures around bundles of fuel rods that help smooth the flow of water past the bundles and through the core.

Similar behavior--but of lesser magnitude than noted in Fukushima 2 was identified in monitoring results at Cooper. Cooper decided to reduce power and flow to ≤ 50%. The NRC directed 10 other BWR's (Arnold, Vermont Yankee, Browns Ferry 1 and 2, Peach Bottom 2 and 3, Pilgrim 1, Brunswick 2, Hatch 1 and Fitzpatrick) which had bypass holes in the core plate to review results from TIP monitoring performed within the last 3 mo to identify any anomalous behavior, and to provide the NRC with the results of the review. A somewhat similar problem is described in I.12. (bjv)

The problem appeared to be generic to the BWR-4 class of reactors only. The BWR-3 class did not have lower core support plate orifices.

The bypass holes were drilled in the core plates to maintain the total bypass flow at ~ 10% of the total core flow because finger springs (FA's were equipped with finger springs on lower core tie plates to provide a controlled flow leakage path between the lower tie plate and the fuel channel) reduced the flow in the core bypass region. Arnold [illegible] dia holes. Plugging the holes w[illegible] only slightly affect steady-state ther[illegible] conditions. The plugs mentioned [illegible] were similar to those previously installed in Pilgrim and Vermont Yankee (see I.12). GE determined that the natural frequency of the LPRM tubes was 2.5 Hz. Among the permanent fixes GE was considering were clasps to hold the TIP tubes stable, diffuser or bubbler pipes to be inserted in the holes in the core plate, drilling 2 holes in the tie plates (transition pieces) at the bottom of the FA's, etc.

Fukushima 2

During Dec 74, an unusual traversing in-core probe (TIP) trace was noted. Evaluation of this trace and subsequent diagnostic tests led to the conclusion there was a high probability that a neighboring channel had developed a hole. The reactor showed 11 to 12% peak-to-peak noise in Feb 75. Prior to the reactor shutdown, a 2nd location developed similar noise characteristics on its TIP traces. Inspection of the channels surrounding these locations at a subsequent outage confirmed the hypothesis. In order to assess the probable cause and amount of damage, a detailed inspection plan was developed and implemented. A total of 197 channels received a detailed visual inspection. The results of this inspeccion are given below.

Type of Incore	Number Inspected	Through Wall Wear	Est. Wear (mils) 30-70	10-30	<10
LPRM	124	3	39	50	32
Source	20	-	8	9	3
SRM	16	-	1	6	9
IRM	32	-	3	11	18
None	5	-	-	-	5
Total	197	3	51	76	67

Cooper

Special instrument checks indicated possible vibrations and they reduced power to ≤ 50% in late Apr 75. Essentially no indications of "noise" or anomalous readings were detected at a 50% power, 44% flow condition. They elected to run at 50% through the summer load period until Oct 75 and then shutdown for a fix. Their refueling was not scheduled until 1976. In July the NRC permitted them to increase power to 60% and reduce flow to 40%.

Arnold

TIP traces indicated 3 possible locations that had higher than average noise levels. Only one location had noise that was not clearly less than reporting criteria established in Apr 75 by the NRC. The TIP traces at Arnold contained less noise than those taken at Fukushima 2.

R In mid-May the NRC ordered them to reduce power to 50% (at core flow rates < ~ 55%, the noise was not observed). The unit was shutdown in early June for an inspection. The shutdown was preceded by a 72 hr test run at power levels and core flow rates up to 100%. They planned to replace all channels which experienced > 10 mils wear in the lower half or > 21 mils wear in the upper half. These limits were consistent with the allowable on-site handling scratches or wear that any channel could experience during normal fuel handling operations and allow ~ yr of additional duty. Inspection of the first 4 channel boxes (adjacent to thimble location with high noise, mentioned above) showed unacceptable wear in the corners adjacent to the instrument thimble. The wear was as much as 40 mils in depth with an additional 8 to 10 mils in several notches in the general wear region. The width of the wear region was as much as 3/8 to 1/2 in. The wear was not as serious as that at Fukushima 2. Altogether, about 1/2 of the channels adjacent to instrument rods showed wear.

They plugged the bypass holes and replaced 119 channels. One FA was dropped during the handling operations. The length of the outage was ~ 6 wk. Plugging the channels reduced the plant capacity by ~ 15%. The NRC then authorized them to operate up to 85% power. In-core instrumentation and accelerometer data obtained after startup indicated there was no impacting of instrument tubes against channels.

Peach Bottom 2 & 3

Unit 3 was derated to 50% power in early June following indications of possible TIP vibrations. Unit 2 had shutdown for local leak rate testing on May 17. After it returned to the line in June it proceeded to 100% power with continuous monitoring for vibrations. It was restricted to 50% power after July 6. Both units were allowed to go to 60% power, 40% flow in late July. Plans were to operate the units at rated power for short periods as part of the fuel preconditioning program. Accelerometers were installed on TIP detector drive cables. Preliminary results indicated there was no contact at < 60% flow. PECO then asked for permission to operate up to 50% flow and 3.62 MWt bundle power. In-core fixes were performed on Unit 2 in Nov 75 and on Unit 3 in Jan 76.

Hatch 1

This unit was able to operate at ~ 80-85% power while monitoring for variations in TIP flux signals. In late Aug they proceeded to 100% recirc flow (~ 95% power) to obtain TIP traces. The TIP values of the ratio of noise band width to signal amplitude did not exceed 6% over any 10 in. of radial core length and they planned to resume their 100% startup testing program to prepare the unit for commercial operations. (Holding the variation in flux signal below 6% allayed NRC concerns about possible chafing and wear from vibration.) During a Nov outage, 192 fuel channels which were adjacent to the 48 in-core tubes were found to have no through wall perforations. However, 125 channels were rejected and removed from service due to significant wear. New channels of the same design replaced those rejected. There were 64 channels identified as exhibiting minor wear and they were relocated in the core in such a manner that worn areas would not be adjacent to in-core tubes. Three channels were found to have no visible wear and were to be returned to their original locations in the reactor core. The bypass flow holes in the lower core plate were also plugged during the outage. Continued monitoring and inspection for vibration induced damage was planned.

Browns Ferry 1

Inspection of all 248 channel boxes adjacent to incore instrumentation was completed in early Aug. The magnitude of the wear distributed among the various types of incore instruments was the same as at other reactors. That is, the total distribution was bimodal and separable on the basis of bypass holes being present or absent. Of the instruments having bypass holes present, the 3 hole locations led the other locations for the magnitude of channel wear. GE had the following 4 inspection result categories: rejectable, probably rejectable, probably acceptable and acceptable. This number of categories anticipated selective channel placement in the core if the availability of channels became limiting. Nine channels had "double wear" and 6 of them were definitely rejectable. "Double wear" was a light surface wear which occurred on the channel box flats adjacent to the worn corner and beyond the wear where channel material had been removed. At least one channel had thru-wall wear. Modifications were authorized in Mar 76 which approved the drilling of the FA lower tie plates of Types 2 and 3 FAs to provide bypass flow. Holes in the lower core support plate orginally provided for bypass flow were to be plugged to eliminate in-core instrument tube vibrations. Units 1 and 2 were not authorized for operation until a later safety evaluation was completed. GE felt that drilling new holes in the lower tie plate assembly would rearrange the coolant flow through the core sufficiently enough to allow operation at 100% power and still prevent instrument tube vibration.

Vermont Yankee

This unit was also able to operate at 80-85% (see Hatch 1 above). They shutdown for 540 hr in Aug and plugged holes. There was some wear.

FitzPatrick underwent in-core fixes in Feb 76 and Brunswick 2 plugged bypass holes in Mar 76. Pilgrim had already plugged bypass holes.
(buj,buk,cne,cnf,cng,cnh,cni)

N Muhleberg

The fuel channels were affected during the first 2 cycles by vibrating poison curtains. This led to a temporary power reduction and subsequent outage (~ 461 hr) to remove some of the curtains and plug bypass flow holes. After removal of the last curtains during the summer 74 refueling outage, a new problem arose; channels were damaged due to vibrating incore detector tubes. This phenomena was hidden "behind the curtains" during the first 2 yr because the poison curtains absorbed most of the energy induced by the leakage flow through bypass holes in the bottom plate. Inspection of all affected channels during the summer 75 outage showed excessive wear (not through holes) on 21 channels. They were replaced and the new cycle was started until a final fix could be determined.

N In early spring 76, preparations were made for a final solution. It consisted of plugging the bypass holes in the core plate and drilling 2 small holes in the fuel bottom pieces. The drilling technique used was a electro-erosion-technique. Tests had been performed at GE and at Muhleberg and the results were promising. The remaining 160 fuel elements were drilled in this manner during the June 76 refueling outage. After some minor start-up problems, all 320 holes were eroded within 16 days (the 80 new elements already had the holes.

Sound emission measurements during start up in July showed that the vibrations were much smaller than in the preceding cycle and that probably no hitting of the channels was occurring. The TIP traces also indicated a substantial improvement. A fuel channel inspection was planned for the 1977 summer outage. (brw, cyq)

See item II.35 for additional indormation.

27. FW SPARGER, NOZZLE CRACKS

Dresden 3 - May 75 - refueling shutdown

Special visual and PT inspections revealed cracks in the FW spargers on 5 of the 8 header pipe to junction box welds and on 1 of 4 thermal sleeve to junction box welds. As with the other plants, the cracking was believed to be due to the design of the spargers which rendered them susceptible to flow-induced vibration.

The program for corrective action was to 1) PT examine the accessible portion of the nozzle blend radius of each FW nozzle, 2) replace all 4 spargers that outage with new spargers to the new design (very similar to the "Design 4" spargers), and 3) inspect the FW spargers during their next scheduled refueling outage. The PT of the FW nozzle blend radii resulted in a total of ~ 150 linear indications being found. The indications were oriented in the direction of the nozzle bore centerline. The max length was 1 1/2 in., with avg length ~ 3/8 in. All linear indications were removed by grinding. The max cavity depth from the clad surface was 3/8 in. The max penetration into base metal was 1/8 in. Grinding was done in 1/16 in. increments and after each increment, PT was conducted to verify whether the indication had been removed. The cross - sectional area of reinforcement removed at the worst grind location was .144 in^2.

The salient design features of the "Design 4" sparger were: 1) interference fit between the sparger thermal sleeve and vessel safe end, 2) Forged-welded tee between the thermal sleeve and the sparger headers, 3) different size and location of exit holes in the sparger, and 4) schedule 80 rather than schedule 40 (304 SS) spargers.

The first design feature was based on several tests showing there was no vibration under any flow conditions when the thermal sleeve was tightly fitted to the safe end (small radial gap). The 2nd feature reduced peak stress levels in the tee by a factor of 4 due to smaller stress concentrations. This was due to the use of full penetration welds, more uniform sections, and large radii at the junction of the header pipes and the thermal sleeve.

The 3rd feature was incorporated to lower the pressure drop at rated flow from 16 psi for the first designs to 11 psi for the new designs which increased the stability margin.

UT and PT of the FW noz[illegible] was planned for Sept 76. They had a tot[illegible] startup/shutdown cycles as of May 76. About [illegible] were used in the sparger replacement. (bos,[illegible],cmx,cvc)

28. IMPROPER TACK WELDING - BROKEN WELDS ON JET PUMPS

Dresden 3 - July 75 - refueling shutdown

During restrainer keeper lift testing of the jet pumps it was noted that several tack welds had parted on the outboard clamp bolt keepers for Nos. 6 and 17 jet pumps. A visual exam by underwater TV had revealed no visible defects on either jet pump keeper; however, during the lift test, which ascertains the resilience of keeper and weld, the welds broke. The apparent cause of both keeper failures was poor tack welding. Properly made, the welds would normally hold the keeper securely to the jet pump restrainer.

Failure of both inboard and outboard clamp bolt keepers could conceivably allow the clamp bolts to back out of the jet pump wedge and restrainer assembly. A double failure of this nature occurring during reactor operation would result in increased vibration of the jet pump and might ultimately leak to separation of the jet pump body at the slip fitting between the mixer and diffuser sections.

The tack weld failures were induced by the artificial vibration of the lift testing. The jet pumps were securely fastened by means of the inboard clamp bolts, both of which were tested satisfacorily. In any event, the routine daily surveillance of jet pump flow characteristics would detect any such discrepancy and the reactor would be shut down within 24 hr.

The clamp bolt keepers were rewelded and successfully retested. These jet pumps were to be inspected during the next major refueling outage to verify the integrity of the tack welds. (bse)

29. CRACKED FW SPARGER

Oskarshamn 1 - Aug 74

The FW sparger was made in one single piece. Cracks were discovered during the summer 1974 shutdown. The replacement sparger was made in segments. The work in the radioactive environment was very time consuming. A special lead container from which the work could be conducted was constructed. This extended an outage by ~ 4 1/2 mo. See item II.38 for additional information. (brw, bto)

3[illegible]. [illegible] LEAKAGE - [illegible]
[illegible]

Pilgrim 1 - Feb 76 - refueling shutdown

Based on the recommendations of GE, NDT of the reactor vessel FW spargers and nozzle inner blend radii was performed to verify their functional integrity as part of the NSSS. VT of the FW spargers was performed and no anomalies were observed. PT of the accessible areas of the reactor vessel FW nozzle inner blend radii was then observed. Therefore, the FW spargers were removed from the reactor vessel and as-built dimensions were obtained so that new spargers could be fabricated. Concurrent with fabrication of the new FW nozzles, all of the fuel was removed from the reactor vessel, and PT was performed on the inner blend radii of all of the reactor vessel FW nozzles. This revealed a total of 78 indications on the FW nozzles inner blend radii surface which extended from a 12 in. reference circle around each nozzle into the nozzle to ~ 2.5 in. past the back edge of the thermal sleeve centering tab. Fracture mechanics analysis was performed and it was determined that the most critical region was the FW nozzle corner.

Corrective action consisted of removing all unacceptable indications by grinding. This repair work was performed in accordance with approved procedures and engineering instructions under the technical direction of GE personnel. As-left grinding repairs were determined by use of etching, dye marking, and molding to provide an accurate sizing of all repair grinding which penetrated the carbon steel base metal material for use in an engineering evaluation. The max grinding penetration into the reactor vessel base metal was 0.5 in.

To prevent recurrence, new FW spargers were installed which incorporated an interference fit with thermal expansion characteristics similar to the surrounding nozzle to reduce the FW sparger thermal sleeve bypass flow, thereby, minimizing flow induced vibrations and thermal cycling.

The FW spargers had been inspected during the 1974 refueling outage and no anomalous indications were observed.

Based on the evaluations made by Teledyne, it was felt that the margin of safety for the structural integrity of the FW nozzles was being maintained at a level required by the original acceptance standards. It was anticipated that a visual reinspection of the blend radii was to be conducted via underwater camera during their next refueling outage. See item XVI. C. 247 for additional information.
(chl,chm,cmi)

31. LOOSE SCREWS - FAILED JET PUMP TACK WELDS SUSPECTED

Vermont Yankee - Fall 1974 - refueling shutdown

During visual inspection of the jet pumps, using a borescope, 2 apparent discrepancies were found. A lock plate screw on jet pump No. 15 and a 1/2 in. retainer cap screw on jet pump No. 6 each appeared to exhibit failed tack welds. The vessel cavity was reflooded immediately after the inspection.

Evaluation of the results of the inspection produced several conclusions:

-GE was concerned that the tack weld failure could possibly indicate a vibration problem that a visual check that had excluded the restrainer wedges could not adequately evaluate.
-Yankee engineering supported GE's position and strongly recommended a completion of the exam including the mechanical tension test.
-The potential hazard presented, in and of themselves, by the 2 small screws, should they become loose in the vessel, was not significant.

A GE engineer experienced with the exam and the jet pump failure at another facility was engaged to supervise the 2nd inspection. The inspection was completely repeated on jet pumps No 1, 6 and 15, and the remaining portions of the inspection were completed on the others. Nothing else of an unusual nature was detected during the inspection. The suspected cracks could not be identified. (cmr)

32. ADDITIONAL FW NOZZLE CRACKS

Millstone 1 - Sept 75 - refueling shutdown

See item II. 20 for previous information. During reexams of the nozzle corners similar PT indications were found. The number of cracks found in the 2nd exam (1yr after the first cracks discovered had been ground out) was greater than the number found in the original exam. The depth of the cracks was about the same. The max depth of the cracks found was ~ .5 in. The max depth of penetration into base metal was ~ .25 in. Less than 50% of the cracks found had penetrated the base metal. The cracks were difficult to detect, even using PT because they were easily plugged by oxide. UT from the external nozzle circumference had not yet shown to be a viable alternative.

The cause of the cracking was still believed to be temperature fluctuations in the nozzle area. Because of leakage around the thermal sleeve, the blend radius area was exposed to temperature swings which ranged from the hot saturation temperature of core water to the relatively cold entering FW temperature. A frequency of 1 Hz could be reached during those swings. Although there was some uncertainty that the cracking was due to leakage flow, the proposed "fix" was based upon the assumption that it was the cause. GE believed that the nozzle area could tolerate, without catastrophic failure, a through-wall flaw under 1000 psig pressure and 100°F FW temp.

GE's recommended interim exam program was based upon the determination of the number of cycles to reach 1/10 of the minimum metal path between the point at which the nozzle is tangent to the blend radius and the exterior surface of the RPV. GE believed that a welded thermal sleeve and sparger, with resultant zero bypass leakage, would be the solution to the problem.

An estimated maintenance time summary was submitted which was applicable to all BWRs with the sparger/nozzle problem. An interference fit sparger design repair would require ~ 12 days and ~ 530 man-rem of exposure whereas replacement would require ~ 10 days and 425 man-rem of exposure. The welded-in sparger design would require ~ 70 days and 2000 man-rem of exposure. Blend radii indication grind-out would require ~ 8 to 20 days and ~ 390 man-rem of exposure. (cms,cmt)

33. FW NOZZLE CRACKS

Dresden 2 - Jan 75, Mar & Apr 76 - refueling shutdown

See item II. 23 for previous information. Upon PT of the nozzle blend radii in Jan 75 a total of ~ 400 linear indications were observed. All indications were removed by grinding. The max grinding cavity depth from the clad surface was 1/2 in. which was equivalent to a depth of 1/4 in. into base metal. The cross-sectional area of reinforcement removed at the worst grind location was 0.481 in.^{2}

Subsequent inspection of the FW nozzles took place in Mar 76. An underwater TV camera inspection of all 4 nozzles was done from inside the vessel as well as UT from outside, utilizing a standard GE procedure for the latter.

Additional exams were performed in Apr 76. A 2nd UT was conducted utilizing the Breda technique which again showed no reportable indications. PT was also performed on the accessible portion of the lower half of one nozzle (240°F). The exam revealed 9 small linear cracks, the longest being 1/8 in. Of the 9 indications, 3 were located in the blend radii of previous grinding. The 2 longest indications were ground out before reaching a depth of .07 in. The remaining 7 cracks were not ground out.

In addition to the PT inspection, UT of all 4 FW nozzles was performed again using the technique developed by Gatti. The inspection showed no indications of reportable magnitude. The results provided further assurance that no significant cracks existed in the nozzles.

Only the lower half of the nozzle was PT examined because the highest stress in the nozzles occurred at the top and bottom; therefore, inspection of either one would provide data concerning the severe cracking. It was estimated that an additional day would have been required to complete a PT of the top half of the nozzle and 5 days and 18 rem to 32 men had already been expended on the job. The UT using the Gatti technique provided assurance that no cracks had propagated into base material 1/4 in. on any of the nozzles, and provided assurance that cracks were not initiating in the root of previous grinding in base material.

Unit 2 experienced 19 start up/shutdown cycles between the inspections done in Jan 75 and Mar 76. The total number of cycles on Unit 2 was 120. It was their normal operating procedure to minimize the number of start up/shutdown cycles and minimize rapid temperature changes in the final FW temp. On the basis of concerns related to the discovery of cracks in the FW nozzle blend radii, GE and the utility were to review operations during start up and shutdown to identify changes in operating procedures which could reduce the magnitude of thermal cycles on the FW nozzles.

Minimum pressurization temps ... would preclude brittle failure with a postulated ... thru-wall flaw were calculated for reactor vessel pressures including inservice hydro testing pressures. The more conservative of the minimum pressurization temps determined from the calculations or the Technical Specs was to be used for inservice hydro testing.

Additional testing of spargers and nozzles was to be performed in late 1976 or early 1977 during a snubber inspection outage. (cmv, cmw, cmx)

34. FW NOZZLE CRACKS

Monticello - Oct 75 - shutdown

During the Sept 75 outage, a crack was found in one of the 4 FW spargers while performing an inservice inspection. They elected to replace all 4 spargers with spargers of a more advanced design. In the process of removing the spargers, cladding defects were observed in all 4 of the reactor vessel FW nozzle corners. PT of the nozzles indicated the presence of ~ 180 cracks. The cracks were "tight" with some cracks penetrating as much as 1/4 in. into the nozzle base material. The cracks were attributed to thermal fatigue of the cladding caused by FW temperature oscillations at the reactor vessle nozzle penetrations. Oscillations of 100°F were estimated at the reactor vessel nozzle corners. The installation of the new spargers, which had an interference fit with the "as left" nozzle, was expected to reduce the by-pass FW flow, which appeared to be the major cause leading to cladding defects.

Repair of the defects consisted of grinding out the cracks. In some instances it was necessary to grind into the base metal to remove defects that had penetrated as much as 1/4 in. into the base metal. PT was repeated after repair to ensure that all defects had been removed.

Confidence in the repair-replacement work was enhanced by amending the PT procedure to increase minimum penetrant and developer dwell times, removing "tired" material from the outer 1/32 to 1/16 in. of clad surface and machining the thermal sleeves of the replacement spargers to final interference fit dimension rather than sizing by cold-work expansion. A total whole body dose of ~ 475 man-rem was distributed over ~ 450 individuals during the sparger replacement. The highest individual dose was ~ 2.5 rem. The initial max radiation levels in the work area were ~ 7 R/hr. (cmf,diu)

35. CONTINUATION OF ITEM II. 26

Cooper - Oct 75 - shutdown

See item II. 26 for initial information. During the outage, all 192 fuel bundles adjacent to the in-core instrument and source tubes were removed and the fuel channels were inspected to determine the magnitude of the wear caused by the in-core tube rubbing and/or impacting against the fuel channel corners. Of the 192 channels inspected, 125 channels or 65% of the total inspected were considered rejects. The reject channels were removed and new channels installed on these fuel bundles. The channel wall was perforated on 4 of the channels.

One perforated channel contained a split in the corner 27 in. in length, several branch cracks and a piece missing leaving a hole in the side of the channel of a size ~ 1 in. by 1 1/2 in. The piece of channel was located in a position flat on the core plate between the fuel support castings at a location previously occupied by the fuel bundle. The piece was retrieved using an underwater vacuum cleaner.

Another channel contained a split in the corner ~ 23 in. in length with several branch cracks.

The 3rd channel contained a split in the corner ~ 21 in. in length, several branch cracks and a piece missing leaving a hole in the side of the channel of a size ~ 2 in. by 3 1/8 in. The piece of channel was located in a position on edge on the core plate between fuel support castings at a location previously occupied by the failed channel. The piece appeared to have dropped directly down from the hole in the channel. The piece was retrieved using a special tool designed and built for removal of the piece.

The 4th channel contained a split in the corner ~ 26 in. in length, several branch cracks and 3 pieces were missing leaving holes in the side of the channel ~ 3 1/2 x 5 in., 2 x 5 in. and 1 1/2 x 1 3/4 in. Two of the missing pieces were located and removed. The remaining missing piece came from the channel hole ~ 1 1/2 x 1 3/4 in. Forty fuel bundles had been removed and/or were removed in the area felt most probable for finding the remaining missing piece. The entire area was carefully scanned by 2 independent inspection teams using an underwater TV camera. Neither team could locate the missing piece. A special crevice device was also used to vacuum around the 9 fuel support castings that had been vacated by the removal of fuel bundles. The piece was not located or retrieved.

From an evaluation of the hole size in the channel, it was believed that the missing channel piece was ~ 0.08 in. thick and roughly the shape of an isosceles triangle with a base of 1 1/2 in. and an altitude of 1 3/4 in. The area of the channel hole was 1.8 in.^{2}. It was felt that the missing piece was located on the core plate. However, since the missing piece could not be located, GE was requested to evaluate the safety implications of the missing channel piece remaining in the vessel. Their evaluation concluded that the 2 areas of concern were interference of control rod insertion and blockage of coolant flow to a fuel bundle. It was highly unlikely that the missing piece could prevent insertion or scram of a control rod. Control rod insertion and scram checks would provide an indication of the problem if one did occur. With respect to blockage of coolant flow to a fuel bundle, it was highly unlikely that the missing piece could attain the required attitude, stay in the required flow velocity paths and continue in a flow stream such that the piece could get below the core plate and be available to block coolant flow to a fuel bundle. In the event the piece did complete such a path, GE concluded that no significant fuel damage would result. If serious fuel damage did occur, main steam line radiation monitors would scram and isolate the reactor. (cnj)

N The lower tie plates of all FA's were to be drilled during the Fall 76 refueling outage. (cit)

36. FAILED JET PUMP RESTRAINER TACK WELDS

Pilgrim 1 - Mar 76 — refueling shutdown

VT of certain reactor vessel interior surfaces with integrally welded internal supports revealed that 9 jet pump restrainer bolt keeper tack welds had separated on 8 jet pumps.

The separations were the result of a small amount of vibration. There was no indication of restrainer bolt keeper rotation. There was no structural degradation of any other part of the jet pumps or reactor vessel components. Since the restrainer bolt did not move and there were no other components affected by the separations, the probable consequences of the event were considered to be minimal (or none). V

A similar inspection of the jet pumps was also performed during the 1974 refueling outage. No indications of reactor vessel jet pump restrainer bolt keeper tack weld degradation were noted during the 1974 inspection. A video tape of 8 jet pumps was also made concurrent with the 1974 jet pump inspection. A review of this videotape was performed and no anomalies were observed.

The preload of a beam bolt was checked at Dresden 2, and found to be acceptable although Dresden 2 also had failed restrainer bolt keeper tack welds and had no indications of major jet pump vibration. It was therefore concluded that tack weld separation was unrelated to preload on the beam bolts. The tack welds of the reactor vessel jet pump restrainer bolt keepers were re-established. (cqf)

37. OPERATING PROCEDURES REVISED TO LIMIT FW TEMPERATURE TRANSIENTS TO MINIMIZE THERMAL CYCLING IN FW NOZZLES

Dresden 2 - 1976

A special study was initiated to determine the corrective action necessary to minimize the FW nozzle crack problem (see items II.27 and 33). It was determined that the probable cause of cracks in the FW nozzle blend area was thermal cycling of the SS cladding. Since the thermal expansion characteristics of the cladding and carbon steel base metal were different, high stresses could result during thermal transients. Eventually, the stresses initiate and further propagate cracks through the clad and into the ferritic base metal.

FW temperature data for the last fuel cycle was selected for review. The largest FW temperature changes ($\geq 100°F/hr$) occurred during plant startups and shutdowns. Out of 18 startups during the fuel cycle, there were 17 startups that had recorded at least one $> 100°F/hr$ event. All of the 18 shutdowns during the cycle had at least one instance of 100°F/hr change in final FW temperature.

1. [illegible]
[illegible]
2-

[illegible] nozzle temperature also underwent significant [illegible]ransients during a "flood-up" of the reactor vessel. [illegible]uring this operation, the water level was brought [illegible]p rapidly to the main steam lines, then slowly [illegible]he main steam lines were filled after which the [illegible]evel was raised rapidly up to the reactor vessel flange, the only limit being the shell to flange ΔT limitation. Since this operation was done through the startup regulating line and the valve cycled (open and closed), the FW nozzle was heated from reactor heat and then cooled from the colder FW. This transient operation caused ~ 4 temperature swings, the avg rate being 280°F/hr with an avg temperature change of 94°F. After evaluating the data, it was concluded that the operations which had the greatest affect on final FW temperature were cutting in and out FW heaters and flooding the reactor vessel.

To investigate a possible method for limiting final FW temperature changes, a special operating procedure was prepared for cutting in FW heaters. This procedure directed the operator to limit the final FW temperature increase to 100°F/hr. This rate of increase was controlled by slowly opening the extraction steam valves and monitoring final FW temperature. This procedure limited the max rate of change of the final FW temperature to 320°F/hr during cutting in of the 1st set of FW heaters and to an overall change of 80°F/hr over ~ 1.5 hr. During previous startups, it was found that heat-up rates of ~ 400-500°F/hour with 90°F temperature changes were common.

The changes in FW nozzle temperatures experienced during a flood-up of the reactor vessel were reviewed to determine possible operational changes to minimize these transients. In light of the reactor vessel to flange ΔT design limits specified in the Tech Specs, it appeard that the method previously described of rapid flooding in steps was the only practical method. The number of flood-ups was to be limited to the extent practicable. (csq)

N GE Service Information Letter No. 208 provided additional recommendations for minimizing FW nozzle thermal cycling. On-off cold FW flow cycling at low power operation was one of the major contributors to FW nozzle thermal duty. When the FW was off, the nozzle heated up to reactor temperature (~550°F), and when the FW was turned on, the nozzle cooled to FW flow temperature (~100°F). This cycling could be minimized on plants with an automatic low flow FW control valve by the proper use and maintenance of this valve. On plants without automatic low flow valves, it was recommended that such valves be added. Bypassing steam to the condenser, should be considered when at low power (provided that the condenser was available), if required to keep the FW flow rate high enough to be controllable (i.e., to avoid "on-off" behavior). Operators should also insure that the FW flow valves and controllers and also the FW heater and drain system valves and controllers were maintained in a condition to achieve quality system performance. Excessive leakage in a full flow FW flow control valve often required FW pump start/stop operation for low flow makeup requirements.

Operating for long peri[illegible] at low FW flow/ temperature should also be [illegible]ized. This could be done by: 1) Perform the [illegible] vessel heat-up at the Tech Spec limit of [illegible] and minimize the total time at low power. 2) Place the FW heaters in service as soon as possible. 3) For BWR's having cleanup return through the FW line, maintain the RWCU system at max flow, and temperature when operating at low power. The warm RWCU flow would mix with the cold FW resulting in higher FW temperature at the FW nozzles. 4) When holding at hot standby, consider operating at less than rated temperature and pressure (e.g. at 350°F and 120 psig). 5) When placing FW heaters and pumps into service, return, if possible, the colder condensate water in the piping and heaters back to the condenser via minimum flow or flush return piping prior to reactor vessel injection. (dly)

38. NUMEROUS CRACKS IN FW SPARGER REQUIRED EXTENSIVE REPAIRS

Oskarshamn 1 - summer 74

See item II.29 for additional information. An inspection of the reactor internals revealed a crack in the FW sparger. Further investigation divulged a number of cracks. The damaged sparger consisted of a continuous ring located at the upper end of the reactor downcomer, about 0.7 m above the reactor core, serving to distribute the FW uniformly in this region. The FW enters the RPV through 6 nozzles near the bottom level of the vessel, flowing up through 6 riser pipes to enter at the bottom of the sparger.

The dose rate in the reactor service room ranged up to a level of ~ 100-300 mrem/hr. Therefore the control station for moving the lead working cage was arranged on the floor below the reactor service room using a transport opening of the building. The reactor service room crane was used to carry the lead cage. All movements of the cage were controlled from the control station and supervised by TV-cameras. The cage was equipped with telephone, TV monitor, electricity, fresh air etc. Normally two men were working in the cage. Conversation and all vital parts of the TV pictures were recorded on tape. The operator was able to observe the working area and handle the equipment outside the cage by using 2 glove boxes on the cage.

To remove the FW sparger ring it was necessary to cut 6 T/C tubes, 6 riser pipes, 6 link brackets and also to cut the sparger ring into 6 parts. A number of special tools were developed. For cutting T/C tubes and pipes, modified standard tools were used. For cutting the FW ring, a plasma cutting robot equipment was developed. The robot also was supplemented with a sludge extraction device. The entire cutting was performed under water. The tools were tested in a full scale model of a section of the FW ring. Testing of plasma cutting was performed also under water. Training of personnel was carried out by using a full scale model of the cage.

The core grid was protected against falling objects by a SS covering sheet. To reduce the overall radiation level in the working area inside the RPV the water level was kept as high as possible. Also special ingots covered with a SS sheet were manufactured and fitted to the front of the cage.

... design of the ... of ... segments. An auto... machine was developed to reduce the radiation exposure of the operator's arms. Extracting of the welding fumes was performed by a fan connected to the active reactor building ventilation system.

With a low water level, the radiation dose rate in the pressure vessel 25 cm above the core grid sheet was 12 rem/hr. By the above mentioned measures the radiation level was reduced to 1.5 rem/hr at the operator's hands outside the cage. The max dose rate inside the cage was 10 mrem/hr, although the normal level was 1 mrem/hr.

The prolongation of the summer shut down amounted to ~ 4 1/2 mo. Only 1 1/2 mo were used inside the RPV. Preparing of tools, methods, shieldings, etc. before the first cutting, demanded 2 mo and clean up inside the vessel 1 mo. The total number of man-hours required was about 74,000. About 120 men were directly engaged in the replacing of the FW sparger. The total full body dose to 107 persons was 43,000 mrem. The max hand dose was 3360 mrem and the mean value of the most exposed hand was 1460 mrem.

(cyq)

39. POISON CURTAIN VIBRATION DAMAGED FUEL CHANNELS

Muhleberg - Aug 73
Vermont Yankee - Oct 73
Pilgrim - 1973

See item I.12

40. FW NOZZLE AND SPARGER CRACKS

Humboldt Bay 3 - July 76 - refueling shutdown

An inspection of the FW nozzle was performed utilizing underwater TV. This inspection consisted of positioning the camera to view most of the circumference of the nozzle as well as the thermal sleeve at the point where it enters the nozzle. The results of the inspection revealed the presence of cracking in the thermal sleeve where it enters the FW nozzle. The cracks extended ~ 3/4 of the circumference of the sleeve. In the region of the 7 to 9 o'clock position on the thermal sleeve (when viewed from the center of vessel) there were several cracks which had resulted in the loss of ~ a 1/4 in. by 1 1/2 in. portion of the thermal sleeve. The thermal sleeve to sparger ring weldment was examined and resulted in the discovery of a crack extending ~ 50% of the circumference of the weldment. The vessel nozzle transition blend radius area exhibited several (3 or more) crack indications in the vicinity of the hole in the thermal sleeve.

The FW sparger ring was also examined and several cracks were observed radiating from the distribution holes on the sparger (max length ~ 1/4 in.). These were assumed to be from thermal fatigue with no further growth expected. Two of the 8 FW sparger support clamps (opposite the side of the thermal sleeve) were observed to be tilted about 10°, but no cracks were found.

Two repair possibilities being considered were installing a new sparger/thermal sleeve design being developed by GE or removing the sparger from the vessel and replacing the thermal sleeve, grinding the nozzle area and returning the sparger to the vessel.

(ddl)

41. LOOSE JET PUMP RESTRAINER CLAMP BOLT KEEPERS, BROKEN TACK WELDS - VIBRATIONAL FATIGUE CRACKING

Dresden 2 - Mar 76 - refueling shutdown

During jet pump inspection, loose restrainer clamp bolt keepers were found on 19 of the 20 jet pumps. Of a total of 40 keepers, 30 were found to be loose as a result of broken tack welds. These keepers were tack-welded to the restrainer assembly to ensure that the gate clamp bolts remained tight. Immediately after this inspection, a tension test was performed on one of the jet pump hold-down beams to determine whether any further slackening had occurred. The hold-down beam tension was found to be 3850 psi; the minimum acceptable tension was 2800 psi.

The original restrainer gate assembly and clamp bolt keepers from jet pump No. 5 were sent to GE for analysis. Each restrainer assembly was found in its proper position, with both clamp bolts fully tightened. The keeper failures had no effect on jet pump operation. Broken tack welds on jet pump restrainer clamp bolt keepers had been found on 2 previous occasions.

In a report entitled "Lab Exam of Jet Pump Restrainer Assembly from Dresden 2," GE stated that the keeper tack weld failures were probably caused by vibrational fatigue cracking. It was conjectured that, with a single tack weld, the keeper tended to be lifted off the gate surface as the result of weld shrinkage. With the keeper thus supported by the weld, jet pump assembly vibrations induced the keeper to vibrate, ultimately fatiguing the weld.

As corrective action, GE recommended that 2 tack welds be placed 180° apart on each keeper. By securing the keeper in this manner, what was termed the "point support mode" would be eliminated, according to the report. Thus the original objective was to reinstall the keepers with 2 tack welds 180° apart. However, difficulties were encountered in operating the welding equipment, in obtaining a suitable welding arc (ground), and in seating the keeper rims. Furthermore, there did not appear to be sufficient accessible keeper rim material to permit the placement of 2 diametrically opposed

... welds. Because of these considerations, the loosened keepers were rewelded to original spec; i.e., one tack weld per keeper. These welds were to be inspected during the next refueling outage, and GE's recommendation would be considered after this inspection. (dir)

42. CRACKED FW SPARGERS AND NOZZLES

Quad-Cities 1 - Jan 76 - refueling shutdown

PT revealed one indication of cracking on each of 2 spargers. The test was being performed upon request by GE, which was concerned over the occurrence of cracking in several FW spargers of the design utilized in Quad-Cities 1 and because of the cracking problems found previously on the spargers in Quad-Cities 2 (see II.25).

On the 60° FW sparger, there was a 1 1/2 in. linear indication which propagated through the junction box-to-thermal sleeve weld. On the 150° sparger, a 2 in. linear indication which propagated through the junction box-to-thermal sleeve weld was located. The linear indications were interpreted as cracks and the inspection was terminated. A work request was initiated to replace the spargers and to inspect the FW nozzles.

Following the removal of the old spargers, PT was conducted on the inner blend radius of the FW nozzles. Numerous linear indications were observed on all 4 nozzles. The majority of these indications were segregated on the upper half of the nozzles. The indications averaged 2 in. in length, and the length of the longest indication was 5 in. These cracks were ground out. On the 60° nozzle, 2 grinding areas protruded a max of 5/32 in. into the base metal. On the 150° nozzle, all grinding was confined to the clad metal. On the 240° nozzle, one grind area protruded 1/8 in. into the base metal. On the 330° nozzle, 7 grinding areas protruted a max of 1/8 in. into the base metal.

The cause of this event was attributed to fatigue of the FW spargers caused by flow induced vibration, and compounded by stresses induced by the thermal gradients inherent between the FW piping and reactor vessel internals. Leakage between the sparger and the FW nozzle contributed significantly to vibration of the sparger assembly and also imposed thermal stresses on the nozzle. The new spargers were designed with an interference fit to eliminate leakage and thus reduce vibration and thermal induced stress cycling. The new spargers were to be inspected during their next refueling outage. (dnm)

43. RADIATION INDUCED ... INTERNAL PRESSURE IN NEUTRON SOURCE ... BERYLLIUM OXIDE ...

Big Rock Pt. - Mar & June 74 - refueling shutdown

Because the analysis of a crud sample obtained during the 1973 refueling outage indicated a .99% beryllium oxide content, an inspection of the initial neutron sources was performed in Mar 74. The attempt to remove the source assemblies from their channels was initially unsuccessful. Only the upper portion of the cylinders and the antimony pins came out. The beryllium-containing cylinders had expanded to the extent that they could not be extracted and were subsequently removed with the channel assembly. The channels were noted to have large cracks in the middle where the beryllium cylinders were located. Both antimony pins were removed and VT with no anomalies observed. One beryllium section was removed from its SS cylinder and was found to be grossly expanded and oxidized.

The failure of the beryllium-containing cylinder was attributed to excessive internal pressures, the subsequent breach of a weld allowing water to enter and resulting in the formation of an oxide accompanied by further vol expansion. Because of relatively close tolerances, the swelling of the cylinders imposed lateral stresses on the source channels resulting in their failure. A limitation on the design life of this type of neutron source was the pressure buildup from helium and tritium gas formation from the (N, α) and (N, 2N) reactions between the fast neutrons and beryllium. This pressure buildup, caused by prolonged use of the beryllium cylinders, might have been assisted by stress corrosion cracking of the 304 SS cylinder.

The SS source channels were replaced by Zircaloy channels. New beryllium-containing cylinders were also inserted, each containing an old irradiated antimony pin and one with a new unactivated antimony pin. The old antimony pins were to be removed when they could be replaced by the irradiated new pins.

In June 74, many loose pieces of material, presumed to be beryllium oxide, were observed in the reactor vessel and small pieces were noted lodged among fuel rods of several bundles. All vessel internals (fuel, channels, control rod blades and incore monitors) were removed to facilitate thorough vacuum cleaning of accessible portions of the reactor vessel. All fuel bundles to be returned to the reactor vessel were cleaned with a water jet. Wedged pieces were dislodged with special tools. The new neutron sources were to be inspected during each refueling outage to insure integrity. (dnd)

N 44. BROKEN NEUTRON SOURCE ASSEMBLY, PIECES IN CORE

N Millstone 1 - Oct 76 - refueling shutdown

While performing a neutron sourse assembly inspection, with all the fuel removed from the reactor, a neutron source assembly was found to be broken. Efforts were made to retrieve the broken source assembly and to locate pieces that were identified as missing. A majority of the broken source assembly was removed from the reactor and the remaining material identified. This material consisted of a 7 in. dia, hollow SS tube, with a .020 in. wall thickness and an inner beryllium sleeve, sheathed in SS with the center portion filled with antimony. The total length of the material was estimated to be 76 in.; 38 in. of the .7 in. dia tubing and 2, 19 in. source [illegible] in. dia. An undetermined [illegible] of this material could be seen on the lower [illegible] support plate. In addition to the lower core support plate inspections, the guide tube areas of the control rods in the area around the broken source were also inspected. One piece, apparently a broken section of the source pin, was located in a control rod guide tube.

To preclude the possibility of the 4 other neutron source assemblies breaking apart, they were replaced along with the broken neutron source assembly with new neutron source assemblies. The exact failure mechanism of the neutron source assemblies was not known. It was also decided to increase the surveillance on the 16 control rods in the 4 by 4 array surrounding the broken source assembly. The normal weekly surveillance of moving each non-fully inserted control rod one notch, in both directions, was to be increased to daily, for the 16 control rods in question for 1 wk of full flow operation. (dwp)

45. BROKEN JET PUMP RESTRAINER CLAMP BOLT KEEPER TACK WELDS

Dresden 2 - Sept 77 - refueling shutdown

During an inspection, Jet Pump #4 outboard restrainer clamp bolt keeper was found with a broken tack weld. Tack welds on all other keepers were intact, with each restrainer assembly properly positioned and both clamp bolts in place. Keeper tack weld failures were attributed to vibrational fatigue cracking. Jet Pump #4 outboard keeper was rewelded with 2 tack welds ~ 180° apart. This would ensure that the keeper could not be lifted off the gate surface during operation. Similar problems were reported in [illegible] and 41. (ebu)

46. FW SPARGER AND NOZZLE CRACKS

Browns Ferry 1 - Nov 77 - refueling
Brunswick 2 - Fall 77 - refueling
Dresden 2 - Sept 77 - refueling
Hatch 1 - Mar 77 - refueling
Peach Bottom 3 - Jan 77 - refueling
Peach Bottom 2 - Apr 76 - refueling

At Browns Ferry 1 the FW spargers and thermal sleeves were removed from all 6 FW nozzles. Machining to remove the cladding and heat-affected base metal from the nozzle was initiated. The interim spargers were procured and were to be installed.

GE had replaced the Brunswick 2 FW sparger with one of a slightly different design. The original installation involved use of a sparger with oversize orifice holes. To correct the condition, orifice plates with the proper diameter holes were tack welded over holes in the sparger to give the proper flow. Exams of the sparger during the Fall 77 refueling showed that several of the tack welds had cracked, thus the sparger was to be replaced.

At Dresden 2, UT was performed on the FW nozzle in Sept 77. There were no recordable indications found in the 4 nozzle bores. One recordable indication was found in the inner radii. This indication was found in the 240° nozzle with an 80% FSH amplitude and determined to be from a deep grind-out cavity left from the 1975 outage. In addition, all 4 FW safe-ends and their associated welds were UT and found to be acceptable.

In Oct 77 the 240° FW sparger was removed. The 240° nozzle inner radius and bore, as well as the accessible areas of the other 3 nozzles, were cleaned with a cleaning solution and very fine (120-180 grit) flapper wheels. The subsequent PT revealed 9 linear indications in the 240° nozzle. There were no indications found in the accessible areas of the other 3 nozzles. Two of the 9 indications were found in the inner blend radius at 5° and 30° and were 5/16 in. in length. These were located in the bottom of previous grind-outs, but were completely removed during the 1st cleaning of that area with 180 grit flapper wheel. The other 7 indications were located in the nozzle bore. They were situated a distance of 5 to 8 inches into the bore at ~180° nozzle azimuth and from 1/8 to 3/8 in. long. Six of these indications were removed by recleaning with the flapper wheel. The remaining indication, the only one which required grinding, was less than 1/16 in. deep. The UT and PT indicated that the interference fit, forged tee spargers had been effective in reducing thermal fatigue cracking as evidenced by the significantly smaller crack depths than would be predicted by generic crack growth models. Dresden 2 had experienced a total of 35 start-up/shutdown cycles since the last complete PT and grinding of the inner radii in Jan 75. The interference fit of the 240° sparger thermal sleeve had remained intact as evidenced by the difficulty with which the sparger was removed and confirmed by measurements taken of the sleeve dia.

Inspection of the FW spargers at Hatch 1 revealed at least 1 fine hairline crack emanating from a hole in one of the junction boxes. Several visual indications which were possibly cracks in the periphery of the flow holes were observed. As a result, they decided to replace all 4 FW spargers. (eal,eam)

During the 1st refueling at Peach Bottom 3 in Jan 77, an external UT was performed on all 6 FW nozzles. UT reflectors requiring further investigation were found on 2 of the nozzles (the "D" and "F" nozzles). The FW spargers associated with the "D" and "F" nozzles were removed, the nozzle surfaces were flapper wheel cleaned, and PT was performed. The PT showed 4 minor indications on the "F" nozzle. No indications were present on the "D" nozzle. The indications on the "F" nozzle were removed by light grinding and did not penetrate the cladding into the base material. VT of both nozzles showed evidence that a metal to metal fit was still maintained at the time of sparger removal (a full circumferential band of unoxidized metal was visible in both safe ends). Both spargers were reinstalled with an interference fit per the recommendations of GE. UT subsequent to the removal of the surface indications showed that the reflectors had not been removed and their characteristics were not changed. Most likely these indications were in the clad bond interface. Even if these reflectors were physical discontinuities, it was highly unlikely that they would propagate due to thermal cycling.

Actions had been taken to mitigate the FW nozzle cracking. The startup procedures had been revised so as to provide increased turbine bypass flow to the condenser during vessel heatup thus providing continuous FW flow during the startup interval. An automatic level control system was installed during the outage to position the 3 in. bypass valve which controlled condensate delivery to the reactor vessel prior to startup of the turbine-driven reactor feed pumps. As pressure increased and the bypass valve approached full open, a reactor feedpump was placed in service with its speed being automatically controlled via reactor level and its discharge valve throttled via operating procedure. The bypass valve control system and the feedpump operating procedure had eliminated batch feeding of condensate and had provided continuous FW flow thus minimizing the number and magnitude of FW temperature cycles.

GE concluded that in light of the results of the nozzle exam and considering that there had been only 7 startups since the refueling outage (42 prior to the outage), there was no undue risk in deferring nozzle surface exams until 1979, assuming no unacceptable UT indications were found in the 1978 exams.

At Peach Bottom 2 in Apr 76, 143 total indications were detected on the FW nozzles using PT. The FW nozzles were recleaned using a flapper wheel and a 2nd PT was performed. A total of 42 indications were observed. The decrease in total number of indications was attributed to the fact that the majority of indications were surface defects in the cladding and could be removed during the flapper wheel cleaning process, prior to the 2nd PT. The 2nd indications were removed by grinding. Only 4 cracks penetrated the cladding into the base metal. All 4 base metal penetrations occurred in the inner blend radius, or in the bore, very close to the blend radius-nozzle bore tangent point. The deepest penetration into the base metal was 1/8 in. (eal,eam,ego,egr,egs,egt,eif)

47. ADDITION TO II.26 - LPRM VIBRATION - FUEL CHANNEL WEAR

Brunswick 2 - Spring 76, Fall 77 - shutdown & refueling

In Mar 76 they shutdown to investigate possible fuel channel damage as reported in item II.26. The program entailed the borescope inspection of 191 of the 192 channels adjacent to in-core monitors during operations. (Bundle BR 444 which was dropped was not inspected for channel wear.) The inspection results showed no through-wall wear on any channels. Eighty-five channels were rejected. This placed the rejection rate at 44.5%, although only 19 channels (10%) had calculated wear > 50% penetration. There were 88 channels (46.1%), which were acceptable for future use in the reactor within restricted locations. Fourty-seven of these channels were placed [illegible] for the continued cycle of operation. [illegible]teen channels had no restrictions on future use.

The last part of the GE solution for reducing the LPRM vibration problems required drilling holes in the lower tie plate of the reload fuel. The new fuel had already been drilled prior to receipt on site. The drilling was performed in the Fall 77 underwater by burning holes through the tie plate with electrodes (E.D.M. technique). Once the holes were burned the assembly was moved to an underwater inspection stand where both the inside and outside of the holes were VT inspected for acceptance. The electodes were replaced after completing drilling runs of about 20 bundles. At the beginning and end of a drilling run, a set of holes were drilled into a replaceable blank in a dummy FA. The FA was lifted out of the water and the blanks removed and inspected again.

Equipment set up was begun by GE ~ 10 days prior to when the drilling operation began following refueling. The drilling stand and inspection stand were located in the fuel pool with the power supplies and control equipment along side on the floor. The equipment was checked out by drilling holes into a discharged FA from the reactor in addition to drilling holes in the dummy FA.

Originally 22 days were scheduled for drilling which was to be done in parallel with the fuel sipping and invessel work. Due to the problems encountered of having to repair some core spray elbows in the reactor vessel the invessel work required much longer than expected and the drilling never became critical path. In spite of repeated drilling equipment failures and some support equipment problems the drilling was completed in 21 days and 19 hr, only 3 hr more than had been scheduled. About 65 hr were lost due to problems with the drilling equipment. The servo valves caused the most trouble, but problems were also experienced with the power supplies and cables. GE had as many as 5 power supplies on site either in repair or in operation and flew parts and equipment in trying to maintain the schedule. (eur,eus)

48. FW SPARGER CRACKS

Gundremmingen - May 76 - refueling

Inspection of the vessel internals revealed cracks in the FW spargers. Exams showed that it was necessary to replace both spargers with new components made of less crack sensitive steel. The cracks were formed by intercrystalline corrosion in material close to welded joints. Because of high radiation levels inside the vessel, the work of replacing the spargers was carried out as deep as possible underwater by divers in a protected "diver bottle". The preparatory work such as procurement of material, fabrication and assembly, shielding, preparations for underwater plasma-cutting, etc. accounted for a large part of the repair apart from the actual replacement operations. The entire operation extended the outage by ~ 2700 hr and resulted in 68.1 man-rem to 89 men. The replacement effort was similar to that performed at Oskarshamn 1 (see II.38). (cyq,euu,evo,evp)

Brunswick 2 - Fall 77 - refueling

See II.46 for previous info. The FW spargers were of the "junction box" design with a SS thermal sleeve. During installation, the FW sparger thermal sleeve was expanded ~ .005 in. oversize and jacked into the FW nozzle. The flow holes on the original FW sparger were initially drilled oversized due to a design error. To correct this problem, orifice plates were tack welded into the incorrect FW sparger flow holes. In the spring of 1976, the FW spargers underwent VT. This inspection revealed several cracks on the flow hole orifice plate tack welds. Weld repair of these cracks was attempted but was unsuccessful.

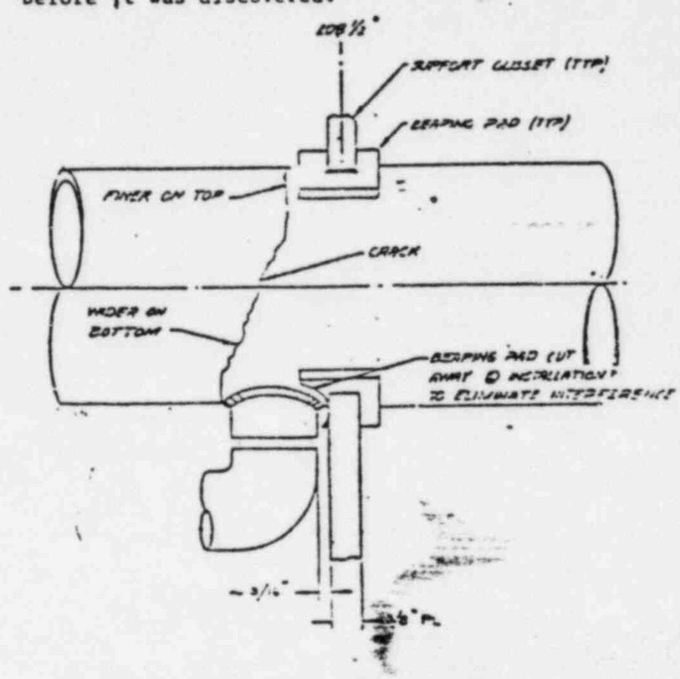
FW spargers with forged tees and Inconel thermal sleeves were ordered from GE. It was decided to use Chicago Bridge and Iron "as built" vessel curvature dimensions for the FW sparger instead of using a template in the vessel. This saved ~ 3 to 4 days critical path time and personnel exposure. The RPV was hydrolased with 6,000 psi water. The hydrolasing was successful and removed almost all of the red oxide. An invessel work platform supported by the core shroud was installed. Lead blankets were hung over the vessel walls and laid over the core spray header to reduce the radiation exposure. The general area radiation in the RPV was 400-500 mR after all shielding was installed. All 4 spargers were removed without any difficulty. They had extremely tight fits in the nozzle and required a special jacking device for removal. After removal, each FW sparger was given a VT. This exam revealed no indications except the flow hole orifice tack welds.

The FW sparger brackets were given VT and no indications were found. Each FW nozzle was prepared for PT by surface grinding with flapper wheels. A special FW nozzle mock-up was used to train each grinder prior to working in the vessel. The FW nozzles were then PT examined. The extent of the PT was 360° around the blend radius and nozzle bore. The entire blend radius from the vessel shell to nozzle forging weld was included. The nozzle bore was examined up to the cladding/base metal interface. The PT was done in accordance with the requirements of the ASME code. Interpretation of the PT exam was made at least 25 min after application. The 45° FW nozzle had 12 indications, the 225° FW nozzle had 5 indications, the 135° FW nozzle had 7 indications, and the 315° FW nozzle had 3 indications. All the indications were small, hair-like and < 1 in. long. They were all removed by light grinding without entering the base metal. FW nozzle leakage land diameters were taken with GE's dial-a-bore. The FW sparger thermal sleeves were then machined .010 in. larger than their respective FW nozzle and welded to the spargers. For installation, the FW sparger thermal sleeves were chilled in liquid nitrogen and inserted into the nozzles. All spargers were installed without any difficulty. Total task time was 162 hr. (eus)

50. CRACKED CORE SPRAY SPARGER

Oyster Creek - Oct 78 - refueling

While performing ISI inside the reactor vessel, an indication was discovered on the core spray sparger for System II. The core spray sparger consisted of 3 1/2 in. schedule 40 type 304 SS pipe formed in 2 semi-circles held in place with brackets attached to the core shroud. Nozzles were welded into the bottom of the pipe about every 5 in. to direct the flow of water directly on the fuel bundles in a pre-established pattern. The circumferential crack (see figure) was ~1/32 in. wide at its widest point and extended ~ 200° around the sparger. It was located at an aximuth location of 208° in the reactor vessel. This was ~ 58° from the inlet and 32° from the end of the sparger arm. The crack was through the wall, as determined by pneumatic testing, and was smaller inside the pipe than on the outside. The crack appeared to have initiated close to one of the spray nozzles and was adjacent to one of the support brackets. Because of the design of the sparger and the mounting brackets, it was concluded that the sparger would have been held in place if called upon for operation even if the crack had propagated completely around the pipe circumference before it was discovered.



The most probable cause of the crack was due to local cold work of the sparger and stresses imposed during installation and fitting of the sparger within the shroud. Fitting the pipe into position during installation could have resulted in stresses large enough to propagate a stress corrosion crack.

The upper core spray sparger repair consisted of the addition of a bracket assembly to provide axial support to the core spray piping in the vicinity of the crack. No attempt would be made to seal the crack. The bracket assembly was constructed of Type 304 solution annealed SS and was held in place by four 3/4 in. bolts that were pre-loaded and locked in place by Class A type locking caps. The bracket assembly was fitted around the existing spray nozzles on both sides of the crack to provide axial support to the core spray sparge in the event the existing

... propagated completely around the pipe circumference.

The addition of the bracket assembly would hold the sparger in such a way that if the crack were to propogate 360°, the max opening that could occur would be 1/16 in. This was based on the clearances of the bracket to nozzles and took no credit for clamping forces. Structural evaluations had shown that the bracket would limit the opening of the crack to $\leq$ 1/16 in. and that the bracket would remain in place. The choice of materials, design, bolt preloading and use of locking caps assured the bracket itself would not come loose. For the bracket to be forcibly removed from the sparger the bracket would have to be deformed at least 1/4 in. and this would require a force of $\sim$ 5000 lb. There was no conceivable mechanism that could apply this type of force to the bracket.

A 360° crack 1/16 in. wide on a sparger would conservatively divert < 2% (61 gpm) of system core spray flow through it. This in itself would not affect the ability of the core spray system to provide 3400 gpm to the core at a reactor pressure of 110 psig, but some nozzle flow would be diverted through the crack. GE calculated that the minimum flow from each nozzle would be at least equivalent to that obtained at 3400 gpm from an uncracked sparger if the system flow was 3700 gpm. In addition, the installation of the bracket would not interfere with the core spray distribution pattern and would not allow water from the crack to directly impinge on spray from the sparger nozzles.

The repair was performed remotely and the final exam to ensure proper fitup was recorded on video tape. The bracket assembly in place was examined at both ends and confirmation of proper fit was made and recorded. (fnq)

... POSSIBLE FW SPARGER AND THERMAL SLEEVE QA DEVIATIONS

... BWRs In General

The NRC conducted an inspection (Sept 79) at the Marvin Engineering Co. to evaluate their overall QA and QC programs. This company was a subcontractor and supplier to GE of BWR internal FW spargers and thermal sleeves. The results established that serious deficiencies existed in the implementation of the QA program relative to the manufacture of these components. During this inspection, 27 deviations from applicable codes, and contractual and regulatory requirements were documented in the areas of material identification and control, process control, welding and NDT exams.

All BWR licensees and construction permit holders were to supply the following information within 90 days for operating plants and 120 days for plants under construction.

1) Determine if reactor FW spargers and thermal sleeves manufactured and/or fabricated by the Marvin Eng Co were purchased and/or installed. Since Marvin Eng was principally a subcontracting company, determine if this equipment originated with the Marvin Co and was eventually supplied through another contractor/supplier. If any was identified, provide a description of this equipment including its purchase date and its design function during both normal and accident conditions.

2) For each piece of identified equipment, provide the performance history associated with its usage, including the cause of any failures or malfunctons and the frequency of such events.

3) Provide information on the suppliers and receiver's QA/QC program in effect at the time of purchase. This information should be discussed in terms of providing sufficient bases for judging that the integrity of the equipment was sufficient to permit plant operation during normal and accident conditions. (gtv)

N52. CS SPARGER PIPING CRACKS

N Oyster Creek - Jan 80 - refueling

While performing ISI on the core spray sparger piping for System II, two indications were discovered. Those indications were subsequently evaluated and determined to be cracks.

The investigation revealed that the first indication was located at an azimuth $\sim$ 152° and adjacent to the sparger header branch weld. It appeared to extend $\sim$ 120° circumferentially. The second indication was located at an azimuth $\sim$ 170°. It ran at an angle of $\sim$ 30°-40° to the centerline of the sparger and was $\sim$ 2-3 in. in length with 2 branch indications running vertically for $\sim$ 2 in.

An air test performed on the sparger (P. F. Avery Corp) indicated that the crack indications might not have been through-wall as evidenced by lack of air penetration. Subsequently, a UT was performed on the areas in question. The first area was located $\sim$ 152° and a through-wall crack was detected in the HAZ of the header branch weld. The second area was located at $\sim$ 170° where additional through-wall cracking was detected.

The cause may have been due to installation methods and was being investigated. (gur,god)

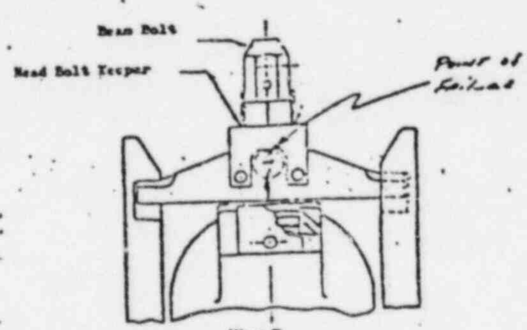
53. JET PUMP BEAM BROKEN, INDICATI...

BWRs in General - 1980
Dresden 3 - Feb 80 - 66% power
Quad Cities 2 - Mar 80 - refueling
Pilgrim - Mar 80 - refueling
Millstone 1 - Oct 80 - refueling

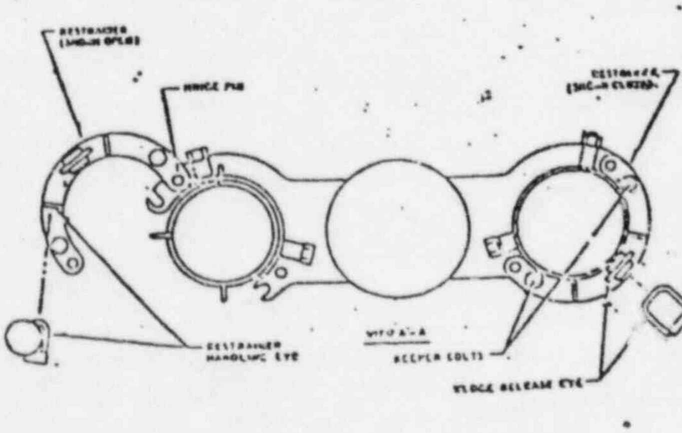
The NRC issued a Bulletin (80-07) regarding BWR jet pump assembly failure after several plants reported jet pump problems.

At Dresden 3, alarms were received in the control room. Observed changes in plant parameters indicated that a jet pump had failed. Shutdown was initiated to bring the Unit to cold shutdown within 24 hr as per Tech Specs. The changes were 1) generator output decrease from 539 to 511 MWe, 2) core thermal decrease, 3) core flow increase from 97.6 to 104.7 x 10^6 lb/hr, 4) core plate dp decrease from 16.1 to 13.8 psid, and 5) "B" recirc loop flow increase from 49 to 54 x 10^3 gpm while "A" loop flow remained at 49 x 10^3 gpm. It was then determined that jet pump No 13 had failed.

Following vessel head removal and defueling, TV camera and VT of the jet pumps and vessel annulus revealed the hold-down beam assembly of the suspect jet pump had broken across its ligament sections at the mean diameter of the bolt thread area. Failure of the beam assembly resulted in pump decoupling at the diffuser connection. Subsequent in situ UT of all other jet pump hold-down beams, using a special UT technique developed by GE revealed ultrasonic indications of cracking at the same location in 6 of the remaining 19 beams examined. A sketch of the typical jet pump assembly is shown in Figs. The indications ranged from 6 to ~ 20 mils. The beams were machined from stock Inconel X-750 material. (hmi)



View B
Jet Pump Beam Details



At Quad Cities 2, a UT was performed on all jet pump hold-down beam assemblies. A width crack indication along the upper machined surface near the beam bolt on jet pump No 16 was discovered. It was estimated in excess of 100 mils in depth and in the same location as found at Dresden.

Pilgrim reported discontinuities on Jet Pump Beams 1, 3 and 15.

The failed assemblies were replaced. A metallurgical analysis was being performed by GE and the cause of the crack was identified as intergranular stress corrosion cracking (IGSCC).

Concerned with the potential for degradation of jet pump function, the NRC recommended the following actions at BWR-3 and BWR-4 facilities:

Plants in their refueling outage, prior to startup, were to: a) assess the integrity of the jet pump structures, the hold-down beam assembly, hold-downs wedge and restrainer assembly, b) conduct UT to assess the integrity of the jet pump hold-down beams at the mid length ligament areas bounding the beam bolt, and c) when startup began initiate the surveillance described below.

Plants in operation were to perform daily jet pump operability surveillance. Individual jet pump dp readings should be recorded and used to establish a data base for expected characteristics for each jet pump. Record and evaluate the following deviations: a) the recirc pump flow differed by more than 10% from the data base for that pump, b) the total core flow was more than 10% greater than the core flow value derived from established power-core flow relationships, and c) the diffuser to lower plenum dp reading on an individual jet pump exceeded the expected characteristics established for that pump.

If it was determined that a jet pump was inoperable or significantly degraded, the reactor was to be shutdown in accordance with Tech Spec requirements. (god,gwo,hbg)

Investigations by GE indicated that beam failure was preceded by a long period of slow crack propagation followed by a short period of rapid crack propagation. During this time, the loaded beam deflects to accomodate the load. Using data from Dresden and from another reactor which earlier experienced similar failure, GE showed that beam deflection during the rapid propagation period allowed significant jet pump leakage. By monitoring loop flow and recirc pump speed during this period, impending beam failure could be detected during the rapid crack propagation period of a week to 10 days before actual failure.

GE studies also indicated that initiation of IGSCC in beams was unlikely before 2.6 yr of operation. After crack initiation, a calculated period of 2.7 yr of crack growth preceded actual failure. Because of such slow crack growth rate, and assuming a reasonable margin of error, UT of all beams during a refueling outage would reasonably assure that no beam failure would occur during the following operating cycle.

GE was in the process of developing a permanent fix to the jet pump holddown beam problem, probably involving reduction of stress on the beam through redesign and modified heat treatment to resist IGSCC, or use of a different material. (hmi)

N Results of VT and UT exams of Millstone 1 jet pump beams indicated that 1 beam was rejectable and 4 beams were in marginal condition. The cause of jet pump beam degradation was believed to be intergranular stress corrosion cracking. The jet pump beams 3, 4, 5, 6 and 7 were to be replaced during the refueling outage then in progress; all were to be inspected and/or replaced during the next refueling outage. Recirc loop flows were to be monitored during the next fuel cycle, as per recommendations set forth in IE Bull 80-07. (isk)

54. SOURCE HOLDERS BROKEN

Peach Bottom 2 - 1977 - refueling

In May, during the removal of source holders, the fourth holder to be removed came apart. The upper section was removed but the lower section remained in the reactor. A TV inspection of the remaining portion, as well as the other 3 source holders, revealed significant cracking of the SS sleeve. A special tool which gripped the holders from the bottom was ordered from the NSSS. The removal of the remaining source holders and the debris left above the fuel support pieces from the broken holders was completed in July. (hao)

55. CS SPARGER CRACKS IDENTIFIED

BWRs in General - 1980
Oyster Creek - 1978, Jan 80
Pilgrim - Feb 80

The NRC issued a Bulletin (No. 80-13) in May regarding core spray cracking reported by 2 utilities (19 of 21 plants inspected observed no cracking).

In 78, Oyster Creek identified a crack in the CS Sparger System II (see II.50). An evaluation postulated that deformation of the sparger had occurred during fabrication and installation which led to cracking by Intergranular Stress Corrosion Cracking (IGSCC). A clamp assembly was installed over the crack and operation continued until refueling In Jan 80, the sparger was inspected and further cracking was discovered (see II.52). A total of 25 cracks, 0.001 to 0.002 in. in width and of varying lengths, were identified in both CS spargers. It was believed that the cracks were undetected during the 1978 inspection due to inspection equipment limitation.

Nine additional clamp assemblies were installed and were expected to maintain the sparger physical integrity. The NRC determined that the repair measures were adequate until an improved replacement system could be installed during the next refueling outage.

An exam at Pilgrim revealed 5 indications in the upper CS sparger and 2 indications on the lower CS sparger. [illegible] varied from 1 to 8 in. and the azimuth [illegible] were 15°, 195° and 345°. The indications were confirmed as cracks after hydrolasing and brush cleaning. An evaluation indicated that the sparger would retain its integrity throughout the next cycle although CS flow distribution could be affected due to through-wall cracks; however, flow delivery to the shroud interior was not expected to decrease. It was believed that a MAPLHGR limit reduction would be imposed during the next cycle to compensate for the assumption of no CS heat transfer. Based on results from other sparger inspections and previous pipe cracking experience, cold work and sensitization during fabrication and installation stresses were considered to be the major factors in causing the observed cracks at Pilgrim. The cracks were hypothesized to be initiated and propagated by IGSCC.

Because the cause of cracking was not yet confirmed by metallurgical analysis, GE was developing tooling to extract sparger samples, and was evaluating methods of improving sparger inspection techniques.

In addition, the NRC informed the BWR facilities to take the following actions:

1) At the next scheduled and each following refueling outage until further notice, perform a visual inspection of the SC Spargers and the segment of piping between the inlet nozzle and the vessel shroud. Remote underwater TV exams were acceptable if adequate resolution could be demonstrated. The viewing in situ of 0.001 in. dia fine wires was considered an acceptable means of demonstrating suitable resolution of the TV examinations. Such techniques as the use of oblique lighting, and the ability to light from each side independently were considered useful in enhancing the image of cracks to facilitate detection.

2) If cracks were identified, the location and extent of the indications were to be reported. Supplementary exams using volumetric methods could be performed to aid in characterizing the extent of cracking in non-visible locations. (hml)

