UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (O.L.)

PREPARED DIRECT TESTIMONY OF
ROBERT N. ANDERSON AND DALE G. BRIDENBAUGH
ON BEHALF OF SUFFOLK COUNTY

REGARDING

SUFFOLK COUNTY CONTENTION 24

AND

SOC CONTENTION 19(c) & 19(d)

CRACKING OF MATERIALS AND

MATERIAL SELECTION

AT SHOREHAM

MAY 4, 1982

SUMMARY OUTLINE OF SUFFOLK COUNTY CONTENTION 24 AND SOC CONTENTION 19(c) AND 19(d)

Suffolk County contends that LILCO has not taken adequate care in the selection and control of materials used in the construction of systems and components making up safety-related systems exposed to the reactor coolant environment. Failure of such materials through the mechanism of intergranular stress corrosion cracking (IGSCC) and through thermal cycle fatigue has been observed in nearly every BWR which has been in service for more than three years. This problem was first observed at BWRs in 1965 and has been the subject of extensive investigation. Pipe cracks have been identified by the NRC as a generic safety issue since that term was first used by the NRC. Two "Unresolved Safety Issues" were finally identified in the NRC's Annual Report to Congress for 1978. These two USI's were A-10, BWR Feedwater Nozzle Cracking, and A-42, Pipe Cracks in Boiling Water Reactors.

The NRC subsequently identified Task Action Plans on these subjects. The Task Action Plans were concluded in 1980 with the issuance of NUREG-0313, Rev. 1, and of NUREG-0619. With the issuance of these reports, the NRC changed the identification of these issues to "technically resolved" but this classification has no effect on the ultimate implementation of the technical recommendations.

LILCO has claimed that these issues are not safety problems but rather represent potential reliability problems for future operation of the plant. Witnesses agree that the potential accidents resulting from pipe ruptures have been analyzed in the FSAR but take the position that safety is potentially affected due to the increased probability of pipe rupture, the potential problem of failure of a major reactor pressure vessel nozzle which has not been analyzed, and because of the safety implications of the increased occupational radiation exposure that may result from required future modifications.

The major concern at the Shoreham plant is the use of 304 stainless steel in the construction of the reactor recirculation system. The recirculation system constitutes the majority of the piping exposed to reactor coolant and is the closest to the reactor core and therefore potentially will attain the highest radiation levels. The 304 stainless steel material has been observed to be susceptible to IGSCC and LILCO has only committed to perform the NUREG-0313 recommended inspection of the non-conforming welds in the recirculation system to the "extent practicable".

With regard to the feedwater nozzle cracking problem, LILCO appears to have committed to make the modifications recommended in NUREG-0619. There is, however, a question as

to when such modifications will be made, the completion of the recommended preservice inspection, and the effectiveness of the low flow feedwater controller that will be utilized.

Witnesses believe that LILCO could and should take further steps to assure adequate performance of these critical materials at Shoreham. Further consideration should be given to improving the as-built condition of the 304 piping and additional leak detection devices should be considered for non-conforming welds. A contingency plan should be developed for the guidance of future action. LILCO's failure to take such steps shows that LILCO has not demonstrated adequate resolution of these important issues in accordance with the appropriate regulations.

Exhibits:

1. U.S. NRC 1979 Annual Report.
2. LILCO May 15, 1981 Letter SNRC-566, J. P. Novarro to Harold R. Denton.
3. FSAR, Section 5.2.3, General Material Considerations.
4. NUREG-0531, Investigation and Evaluation of Stress Corrosion Cracking, Section 6.0.

PREPARED DIRECT TESTIMONY OF
ROBERT N. ANDERSON AND DALE G. BRIDENBAUGH
REGARDING SUFFOLK COUNTY CONTENTION 24
AND SOC CONTENTIONS 19(c) & (d)
CRACKING OF MATERIALS AND MATERIAL
SELECTION AT SHOREHAM

I. INTRODUCTION

1. This testimony was jointly prepared and edited by Dr. Robert N. Anderson and Dale G. Bridenbaugh.[1/] A statement of the qualifications of Messrs. Anderson and Bridenbaugh has been separately provided to this Board. With regard to this particular contention, the experience of Mr. Bridenbaugh is particularly relevant. He was involved for five years in the construction, startup, and maintenance of the Dresden 1 plant (Commonwealth Edison). Shortly thereafter he took over as manager of General Electric service for operating reactors and was responsible for the investigation and repair efforts for the pipe cracks experienced at Dresden 1 in the late 1960's. In this assignment he was also responsible for directing G.E.'s efforts in the correction of IGSCC failures at the Nine Mile Point plant, the Garigliano plant in Italy, and at Tarapur in India. When the increased incidence of pipe cracks at operating reactors was observed in 1974, he was responsible for coordinating

1/ The initial outline and preliminary draft of this testimony was prepared by D.G. Bridenbaugh, with the exception of Section III.D, which was drafted by Dr. Anderson. Subsequent editing and modification was a mutual effort.

the General Electric Nuclear Energy Division response to the NRC's shutdown order in 1974. Since leaving General Electric, he has continued to follow this generic problem and has submitted testimony in the Black Fox construction permit case and has worked on two studies of this problem for the Swedish Nuclear Inspectorate. Dr. Anderson is eminently qualified in the field of materials, has served as a consultant to the Electric Power Research Institute, and to the California Energy Commission. Dr. Anderson is a Professor at San Jose State University and teaches graduate courses in Corrosion Engineering.

II. STATEMENT OF CONTENTION

2. The purpose of this testimony is to address Suffolk County Contention 24 and SOC Contentions 19(c) and (d) as admitted by the Board as follows:

SC 24: CRACKING OF MATERIALS

> Suffolk County contends that LILCO has not demonstrated, and the NRC Staff has not verified, that Shoreham meets the requirements of 10 CFR 50, Appendix A, GDC 4, 14, 30, and 31, with regard to the adequacy of material selection and control and system design as follows:

(a) The use of appropriate materials and processes as specified by NUREG-0313, Revision 1, has not been fully followed in the design and construction of the Shoreham piping systems important to safety.

(b) Recommendations contained in NUREG-0619 (p. C-12) relating to the installation of a low flow controller to be used to control feedwater flow over a range of flow from 0.5 percent to 10 percent of rated flow has not been implemented at Shoreham. Analytical evidence shows that such a flow controller is necessary to limit crack growth in BWR feedwater nozzles over the life of the plant.

SOC 19(c): REG. GUIDE 1.31: FERRITE CONTENT IN STAINLESS STEEL

(c) Regulatory Guide 1.31 -- The control of ferrite content in stainless steel weld metal by LILCO complies with Revision 1 of the guide rather than Revision 3, with regard to verification of delta ferrite content of filler materials and to examination for ferrite content by a magnetic measuring instrument. Therefore, Shoreham does not comply with 10 CFR Part 50, Appendix A, Criteria 1 and 14.

SOC 19(d): REG. GUIDE 1.44: MATERIALS

(d) Regulatory Guide 1.44 -- LILCO has not adequately demonstrated control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking as required by 10 CFR Part 50, Appendix A, Criteria 1 and 4 and 10 CFR Part 50, Appendix B in that Shoreham has failed to comply with the NRC Staff position described in NUREG-0313, Revision 1 as follows:*

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* It should be noted that LILCO's use of alphanumeric designations is not in compliance with the nomenclature set forth in NUREG-0313, Revision 1. This discrepancy makes LILCO's commitment inscrutable.

1. Portions of the reactor recirculation system (B31) and stainless steel to carbon steel transition welds between the reactor recirculation system and the reactor water clean-up, core spray, and residual heat removal systems do not meet the guidelines set forth in NUREG-0313, Revision 1, for ASME code class I and II reactor coolant pressure boundary piping.

2. The commitment to inspect portions of the reactor recirculation system and transition welds that have been classified as "non-conforming" per NUREG-0313, Revision 1, has been conditioned by LILCO to be limited "to the extent practicable" due to physical interference in some locations. NUREG-0313 does not specifically allow for such deviations. Also, LILCO has failed to identify specifically the number, location, and detailed justification for these deviations. Further, LILCO's objection to the "service sensitive" classification of recirculation riser lines and inlet lines at the safe-end curves demonstrates a failure to comply with the requirements of NUREG-0313.

3. The limiting conditions for leakage included as part of the technical specifications has not been demonstrated in that the leak detection system may not adequately enhance the discovery of unidentified leakage as required by NUREG-0313.

The results of our review of some of the important matters encompassed by these contentions are summarized in the following paragraphs.

III. DISCUSSION OF ISSUES

III.A.: Background and Summary of Position

3. The basic concerns expressed by these contentions deal with the potential inadequacies of the primary system materials, particularly their selection and control, to assure retention of the reactor coolant pressure boundary (RCPB) and

of the safety-related piping systems' integrity for safe plant operation. Cracking of stainless steel materials used in critical piping systems has been a generic problem with boiling water reactors since the mid-1960's. Cracking of a bypass line in the Dresden-1 reactor recirculation system was observed in 1965 and investigation of this problem and work to develop suitable alternate materials has been an industry wide concern for the past 15 years. BWR piping system cracking was identified as a generic safety issue on the NRC's first generic issue list issued in 1977, and in 1978 this issue was reported by the NRC to Congress as one of the 17 unresolved safety issues affecting existing and future boiling water reactors. With respect to the Shoreham Operating License application and this contention, two unresolved safety issues were identified by the NRC which are of relevance. These are:

A-10, BWR Feedwater Nozzle Cracking and

A-42, Pipe Cracks in Boiling Water Reactors

Copies of the summary sections of the 1978 NRC Report to Congress identifying these issues as unresolved safety issues are appended as Exhibit 1.

4. Following the NRC's identification of the A-10 and A-42 unresolved safety issues, these problems were given top priority within the NRC and technical solutions and/or improvements were developed and documented in two NRC reports.[2/]

2/ NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", July, 1980 and NUREG-0619, "Feedwater Nozzle and Control Rod Return Line Cracking", November, 1980.

These reports, issued in 1980, formed the basis for the NRC to declare these problems as "technically resolved", and this position has been so documented in the NRC's report series on unresolved safety issues.[3/] A number of points must be emphasized with regard to this "technical resolution":

(1) The technical advances were not made in the confines of the NRC's Task Action Plans. The NRC's Task was limited to the review and approval of material development that result now from years of nuclear industry research and development. NUREG-0313, Rev. 1 formally recommends implementation of improvements that have been under investigation for a number of years.

(2) "Technical resolution" means only that the NRC believes the technology exists to resolve the problem and makes no judgment as to whether it has actually been implemented at existing plants.

With respect to implementation, NRC plans were that commitment letters would be solicited from each licensee with respect to the two issues and that these letters would be reviewed by the Staff and decisions would be made as to their adequacy. A letter with respect to these two issues has been submitted by LILCO to the Staff (See LILCO letter SNRC-566, dated May 15, 1981. A copy of this letter is appended as Exhibit 2.) As will be discussed in later sections of this testimony, the commitments made by LILCO do not assure complete adherence of the Shoreham Nuclear Plant to the recommendations contained in the NRC's implementation documents.

3/ NUREG-0606, Vol. 3, No. 2, Unresolved Safety Issues Summary, May 15, 1981.

5. An additional discrepancy in the LILCO material program is failure to include improved methods in the construction of the plant. One such improvement is use of advice contained in the latest revision of Regulatory Guide 1.31, Ferrite Content in Stainless Steel. Our opinion is that the failure to fully implement the improvements identified in these three documents (NUREG-0313, NUREG-0619 and Regulatory Guide 1.31, Rev. 3) represent deficiencies that result in the plant not being in compliance with the appropriate regulations; namely, 10 CFR Part 50, Appendix A, Criteria 1, 4, 14, 30, 31 and that they further serve to increase the risk to the public health and safety.

III.B: Industry Experience

6. As indicated in previous sections of this testimony, the problem of pipe cracks in safety-related piping systems in boiling water reactors has been a serious problem nearly since the beginning of BWR commercial reactor operations. Even prior to the identification of this issue as an unresolved safety issue by the NRC in 1978, numerous failures experienced at BWR's, both inside and outside the United States, had caused the NRC to activate a task force as early as 1974 to study the potential safety impact of these failures. This task force, known as the Pipe Crack Study Group (PCSG), was formed by the

NRC following a number of failures experienced in 1973 and 1974. The PCSG published its findings in a 1975 report, NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants", published October, 1975. In additon to the PCSG effort, all BWR's in the U.S. were required to be shut down in late 1974/early 1975 for mandatory piping inspections to assure that undiscovered cracks did not exist. The basic finding of the first PCSG review was that intergranular stress corrosion cracking (IGSCC) is a relatively complex, progressive cracking mechanism that occurs under the simultaneous presence of three conditons: high stress; sensitized material; and a corrosive media. In the BWR application, welding of the commonly selected piping materials provides the stress and the sensitization, and the relatively high oxygen levels inherent in the BWR coolant provide the corrodent. This early PCSG report also found that this issue, although a serious reliability problem, was not considered to be a safety problem because experience to that time had favored the so-called "leak before break" theory.

7. In June, 1978, a through-wall crack was discovered in an inconel recirculation riser safe end at the Duane Arnold Plant. Discovery of this leaking crack was the result of a routine walk through inspection of the drywell following a shutdown for other reasons. The leak detection system had not given

indication of a leak during operation. Through wall cracks were found in the 10 inch riser and extended around 270 degrees of the piping circumference. Furthermore, it was found that cracking was present in the other seven recirculation risers in the same location. This event gave rise to concern over the random event theory ascribed to such cracks and also brought question to the ability of the leak detection systems in use to detect cracks before potential full rupture of the pipe itself. A second event observed at a German BWR designed by General Electric was the discovery of sensitized safe end cracking in large diameter (24") pipes. The combination of these two events caused the NRC to reactivate the PCSG in September, 1978. The PCSG studied the issue for approximately one year and its findings were published in a 1979 report, NUREG-0531, "Investigation and Evaluation of Stress Corrosior Cracking in Piping of Light Water Reactor Plants", published in February, 1979. The data assembled during the PCSG investigation formed the technical basis of NUREG-0313, Rev. 1.

8. In addition to the piping failures experienced at operating BWRs reported in NUREG-75-067 and NUREG-0531, other material failures have been experienced at U.S. BWRs These include cracking of vessel feedwater nozzles, cracking of internal feedwater spargers, cracking of control rod drive return nozzles, and cracking of a core spray sparger. Details

of the vessel feedwater nozzle and control rod drive return line nozzle cracking are contained in an NRC report, NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking", published in July, 1977. The feedwater nozzle cracking that has been experienced at almost all U.S. BWRs which have been in operation several years is of the most significant concern. Safety analyses supporting the licensing of BWRs do not consider the possibility of vessel rupture. Cracking of the feedwater nozzles, an integral part of the vessel, raises questions on the wisdom of such assumption. The possibility of such a crack growing into a vessel rupture case, the extreme difficulty of repairing such defects, the high radiation exposure associated with the repairs, and the possibility that irreparable cracks could be generated which would result in the need for early retirement of the unit serve to emphasize the safety significance of these material failure issues. Correction of this problem (feedwater nozzle cracking) was the subject of intensive work by the industry and by the BWRs designer, General Electric. The NRC report, NUREG-0619, presents the latest state of technical knowledge in this problem area and recommends usage of the latest design thermal sleeve (a triple sleeve) plus other operating and inspection procedures. LILCO has committed to utilize the triple sleeve feedwater sparger design at Shoreham,

but LILCO's documentation does not state if this modification will be completed before initial operation. LILCO also commits to the installation of a low flow controller, but waffles on whether this controller will actually meet the requirements specified in NUREG-0619.

9. Regulatory Guide 1.31, Ferrite Content in Stainless Steel, was issued to assure proper control of ferrite in stainless steel weld metal as installed. The latest version of this Regulatory Guide, Revision 3, requires verification of delta ferrite content of filler materials to be measured by a magnetic measuring instrument. This would provide additonal assurance that the as-welded conditon is as desired and of less susceptibility to cracking. LILCO, however, has committed in the FSAR only to comply with Revision 1 of the Guide which does not require this additonal step. In view of the demonstrated susceptibility of boiling water reactor piping systems to numerous cracking failures, we believe it is essential that the latest technical advances be utilized and that this additional test should be performed.

III.C: Shoreham Commitments

10. LILCO commitments with regard to assuring compliance with the latest technical recommendations for the RCPB and safety related piping systems in the use of crack resistant materials

is contained in Section 5.2.3 of the FSAR and is supplemented by LILCO commitment SNRC-566. The position stated in the FSAR is as follows:

> "The use of severely sensitized or furnace sensitized stainless steel is prohibited. However, exception is taken to Regulatory Guide 1.44 which effectively prohibits the use of welded pipe 304 stainless steel by considering it the same as severely sensitized material and by placing limits on oxygen concentration which cannot be met by operating BWR's." (A copy of this FSAR section is appended as Exhibit 3).

LILCO supplements this position in the May 15, 1981, letter SNRC-566 and states that "the problem of stress corrosion cracking has, to a large degree, been eliminated for Shoreham". LILCO points out in this letter that the recirculation bypass line has been eliminated, that the core spray line and safe end materials have been changed and the CRD return line has been eliminated. This then leaves the reactor recirculation system as the major primary pressure boundary piping which utilized 304 stainless steel material.[4/] LILCO states that modification of the piping material for this system to an alternative material was not practical in the time table for Shoreham. Unfortunately, the 304 reactor recirculation system represents by far the majority of the piping (in terms of internal surface exposed to reactor coolant) and, therefore, still provides a

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4/ LILCO's submittals in repsonse to NUREG-0313, Rev. 1, are silent on RHR/LPCI, and auxiliary piping system connection materials. These systems admittedly are a small percentage of the RCPB when compared to the recirculation system, but their exact material status is not identified.

major opportunity for the initiation of IGSCC failures. We have not attempted to exactly quantify the piping system percentage of the wetted area provided by the recirculation system. It should be emphasized, however, that the recirculation system is the largest and most complex piping system making up the nuclear system. NUREG-0531 contains a good summary description of the piping systems making up a BWR. A copy of Section 6.0 of this report is appended as Exhibit 4. Note in particular Figure 6.1 which shows the complex discharge header and riser configuration. Section 6.2.2 compares the number of welds contained in a typical BWR recirculation system to the number in the core spray piping. The recirculation system contains 132, while the core spray system has 54. However, if the pipe diameters are taken into account, we find that the recirculation system contains about three times the lineal footage of pipe welds that the core spray system does. LILCO attempts to justify the 304 piping in the recirculation system by claims that field weld improvement measures were utilized, including welding process control, grinding restrictions, and weld filler material ferrite control as required by Requlatory Guide 1.33.[5/]

5/ Regulatory Guide 1.33 is Quality Assurance Program Requirements, Operation, so LILCO in all probability was referring to Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal. LILCO's apparent uncertainty in exactly which Regulatory Guide they followed and their failure to identify the revision number (there are three revisions of 1.31) does little to assure confidence in the program as conducted at the plant.

LILCO further commits in SNRC-566 (Exhibit 2) that recirculation system welds and transition welds classified as "non-conforming" will be inspected in accordance with Part III.B.1.0 of NUREG-0313 to the extent practicable for the Shoreham Nuclear Power Station (emphasis added).

11. LILCO's commitment with regard to feedwater nozzle hardware is also contained in SNRC-566. This letter states that the General Electric triple sleeve sparger has replaced the welded-in design. It also states that performance of a baseline UT will be performed after installation of the sparger. However, the Safety Evaluation Report (NUREG-0420, Supplement 1) states the Applicant has committed to install the triple sleeve sparger. SNRC-566 further states that the baseline inspection will be performed after the installation of the sparger. No date is stated for sparger installation. While there is no doubt that LILCO intends to install the new sparger, it is not clear that it has yet been done, nor that it will be done before plant operation. LILCO also states that a low flow feedwater controller (0.1% capability) has been installed, but that monitoring at startup could determine a need for "additional controls." This implies the use of procedural controls. This does not meet the guidelines of NUREG-0619, which call for the use of an accurate controller.

III.D: Material Selection Uncertainties

12. Stainless steel type 304 piping was specified in early BWRs as the material to be used in most of the safety-related piping systems exposed to reactor coolant. This material was selected for this application until very recently, partially because of its common availability and also because it was thought to be an acceptable material for this application. Cracking (IGSCC) experienced over the past eight years had led to the conclusion that 304 is unsatisfactory for use in BWR coolant systems unless it can be left in the so-called unsensitized condition so as to guard against cracking and failure. Some improvements have been made in installation techniques, but the field welding process makes total control nearly impossible. The following paragraphs in this section (III.D) further explain the nature of the 304 failures and demonstrate that the uncertainties in prevention and early detection of failures makes replacement of this piping with a more suitable material desirable.

13. Experience and studies to date have shown that the combination of three conditions, high stress, sensitized material and a corrosive media, will produce IGSCC. IGSCC of austenitic type 304 stainless steel originates in the heat affected zone (HAZ) adjacent to a weld. The welding operation causes heating and cooling of the metal through the range of 870-425°C which, in turn, produces chromium carbide precipitate at grain boundaries. This precipitation reduces the chromium content available for

resisting chemical attack at grain boundaries. These altered grain boundaries are referred to as sensitized because corrosion can penetrate deeply and quickly along the chromium depleted boundaries if high stresses and an aggressive environment are present. Since all three of the above mentioned conditons must be present to some degree for IGSCC to occur, it is theoretically possible to avoid its initiation by eliminating one or more of the three conditions.

14. Avoidance of sensitization can diminish or minimize the onset of IGSCC by using low carbon (less than 0.04 weight percent carbon) material. Another successful method has been the use of corrosion resistant cladding on the inner surfaces of the 304 pipe material so as to isolate the sensitized regions from the aggressive environment.

15. The elimination of high stresses or optimal control of stresses is the second possible solution to the problem. Relieving of residual stresses caused by the non-equilibrium cooling after welding can be done by the use of heat sinks on the interior of the pipe during welding or by induction heating after welding. There is, however, no practical means at the present time to verify piping stress reduction to a sufficiently low level to conclusively prevent IGSCC from initiating in 304 material.

16. The third potential area for solution of this problem is to control the environment under which the material

operates so as to remove the corrosive media. Dissolved oxygen in the BWR coolant at a level of 0.2ppm can act as a corrosion catalyst if sensitization and tensile stresses are present. The presence of oxygen in BWR water is an inherent characteristic of the process and control of it to the degree necessary to protect the sensitized pipes does not appear to be feasible at this time.

17. Certain procedural controls can serve to retard the onset of IGSCC. Limiting grinding on the inner pipe surfaces and prevention of microfissures by control of ferrite in the weld helps to prevent crack nucleation. Unfortunately, the efficacy of such controls cannot be readily verified, and the safest approach remains to use nuclear grade materials that are not readily susceptible to IGSCC.

18. Numerous studies have been performed and continue to be under way in an attempt to identify the best alternate material to use so as to minimize or prevent IGSCC.. Studies have identified a number of materials that are more suitable in BWR applications where conditions conducive to the promotiom of IGSCC are present. One material that has been identified and used is 304L (low carbon) stainless steel. This material is commonly used in industry for such service and has been used as replacement material for many of the failures experienced at BWRs to date. 304L has 85% of the yield strength of 304 and can be substituted directly for 304 in most applications. Its primary drawback is

increased cost. In the past it has generally been less available in the commercial market but that situation appears now to be changing quite rapidly.

19. Certain of the studies conducted in recent development programs have focused on the development of techniques to measure 304 sensitization so as to permit identification of as-installed piping systems that will be more susceptible to IGSCC. The electrochemical potentiokinetic reactivation (EPR) method is one that appears promising. The EPR method measures the current flow during a potential sweep and compares this measurement to a similar measurement of flow in non-sensitized material. The results of this measurement correlate to the actual rate of corrosion in the grain boundary when corrected for grain size. The EPR method appears to effectively identify sensitized materials and is useful for quality control but it has not been proven effective in the field construction application where electrolyte, stress level, and grain size are difficult to control. It is possible that EPR measurement could give some indication of the liklihood of IGSCC at Shoreham, but a conclusive outcome from such a test is unlikely.

20. Testing for the presence of IGSCC in existing piping is difficult because of the nature of the cracks. The cracks are small and tight with many branches and ultrasonic testing

and radiography techniques have some problems in the unequivocal identification of cracks. Moreover, once detected, crack size and hence, severity, may not be readily identified. Acoustic emission using the sound of deformation or cracking in metal is one possible detection technique, but this method is hampered by the problem of background noise.

21. Two arguments that have been advanced to support the use of sensitized 304 stainless piping in reactor application such as the recirculation system are (1) the leak-before-break argument which assumes that IGSCC will propagate asymmetrically and exit the pipe to produce a leak before producing a significant loss of strength in the pipe. The leak then would be detected before the pipe crack propagated far enough to leak in the pipe where it fails under normal service loads and (2) smaller pipes are more likely to fail by IGSCC than larger pipes. The first argument ignores the very real possibility that multiple cracks may occur in the same HAZ which would rapidly weaken the pipe before any signs of leaking might occur. Moreover, some experts have pointed out as quoted in NUREG-0313, Rev. 1 that:

> "The residual stress distribution in circumferential welds tends to promote cracking all around the circumference. As the crack extends around a significant portion of the inside wall, the residual stresses in the axial direction should increase and accelerate the crack growth." 6/

6/ NUREG-0313, Rev. 1, "Comments" by Lawrence Livermore Lab.

Finally, investigators have stated that differences in the susceptibility to IGSCC between large diameter and small diameter 304 pipe may be due only to heat-to-heat variations in the material. It is clear that crack propagation rate and the assumption that crack growth is asymmetrical with the walls of the pipe are only hypothesis at this point and the leak-before-break argument can be considered a shibboleth.

IV. CONCLUSIONS

22. LILCO's primary reason for retention of the 304 material in the Shoreham recirculation piping appears to be that the cost of such modifications would be excessive and that changing of the system to an alternative material "was not practical in the time table for Shoreham".[7/] In our opinion, the continued use of type 304 stainless steel with its limited life, difficulty of detecting failure mode, and the possibility that repairs will exacerbate the cracking means a commitment to future detection and repair that can have adverse consequences for the safety and economics of the Shoreham station. It is for these reasons that susceptible piping should be replaced with material that will be less adversely affected by the oxygenated water. We recognize, however, that these piping systems are already constructed, that the cost and schedule delay associated with full

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7/ LILCO letter SNRC-566 (see Exhibit 2).

replacement would be substantial and that it may be impractical or impossible to effect such a change at this time. We believe that those sections of recirculation system piping that may be considered as "high-risk" weldments that cannot be replaced should be equipped with state-of-the-art (solid state) leak detector elements in the welded areas and backed up by appropriate acoustical detectors when available. The "high-risk" weldments should be identified through the use of a careful study of piping and electrode material composition, review of stress levels at each joint, and augmented by utilization of EPR and delta ferrite measurements.

23. LILCO further should identify all welds in these systems which are not fully available for ASME Section XI in-service inspection. Special attention should be given to leak detection at these joints. We also recommend an evaluation of accidents resulting from possible failures at these locations to supplement hazard analysis already contained in the FSAR.

24. Based on our review and discussion summarized in SEction III above, we conclude that:

(1) LILCO has not demonstrated full compliance with NUREG-0313, Rev. 1 and, accordingly, does not meet 10 CFR 50, Appendix A, Criteria 1, 4, 14, 30 and 31. Deviations include extensive use of 304 stainless piping materials and

failure to apply augmented inspection to all service sensitive applications.

(2) LILCO has not complied with Regulatory Guide 1.31, Rev. 3. Compliance with this Regulatory Guide could add assurance of quality construction of the critical piping systems.

(3) LILCO has not committed that the triple-sleeve sparger as called for in NUREG-0619 will be installed on a timely basis. They have further not committed that an effective low feedwater controller will be provided at Shoreham.

25. We recommend that:

(1) The requirements contained in NUREG-0313, Rev. 1 should be fully followed at Shoreham. This means that the 304 material in the in the recirculation system should be replaced with an alternate material or all welds in the system should receive appropriate post-weld treatment or should be protected from the coolant environment.

(2) If recommendation (1) above is not fully completed with, LILCO should be required

to fully justify all deviations from the NUREG document. Such justification should include:

(a) Issuance of a full report, including isometric piping drawings showing the locations of all 304 piping systems exposed to reactor coolant. All weld locations should be indicated as should all areas where access or other conditions prohibit full compliance with ASME Section XI inspection requirements.

(b) A feasibility study should be performed to determine if any of the above identified piping can be made more resistant to the onset of IGSCC through the use of CRC, IHSI, or other post-installation methods.

(c) A contingency plan should be prepared addressing the cost, schedule, and availability of alternate piping material in the event of post-operational failure of the 304 stainless piping systems. The plan should also quantify

occupational radiation exposure required for the complete replacement in the future, at various times in plant life.

(d) LILCO perform measurements of deposited weld metal to bring the as-built systems into compliance with Reg. Guide 1.31, Rev. 3, to the extent possible.

(e) Apply the EPR test to the installed system to the extent possible to obtain the possible benefit of this test.

(f) Investigate the use of advanced leak detection methods such as the high-sensitivity leak tapes (these are now being developed under an NRC contract) to ensure adequate detection at all non-conforming joints and locations.

(g) Document and make available a summary report showing that all 304 stainless piping material certificates verify the level of carbon and cobalt material in the RCPB piping systems. Utilization of low levels will demonstrate lower probability

of failure and reduce the likelihood of higher radiation exposure rates.

We additionally recommend:

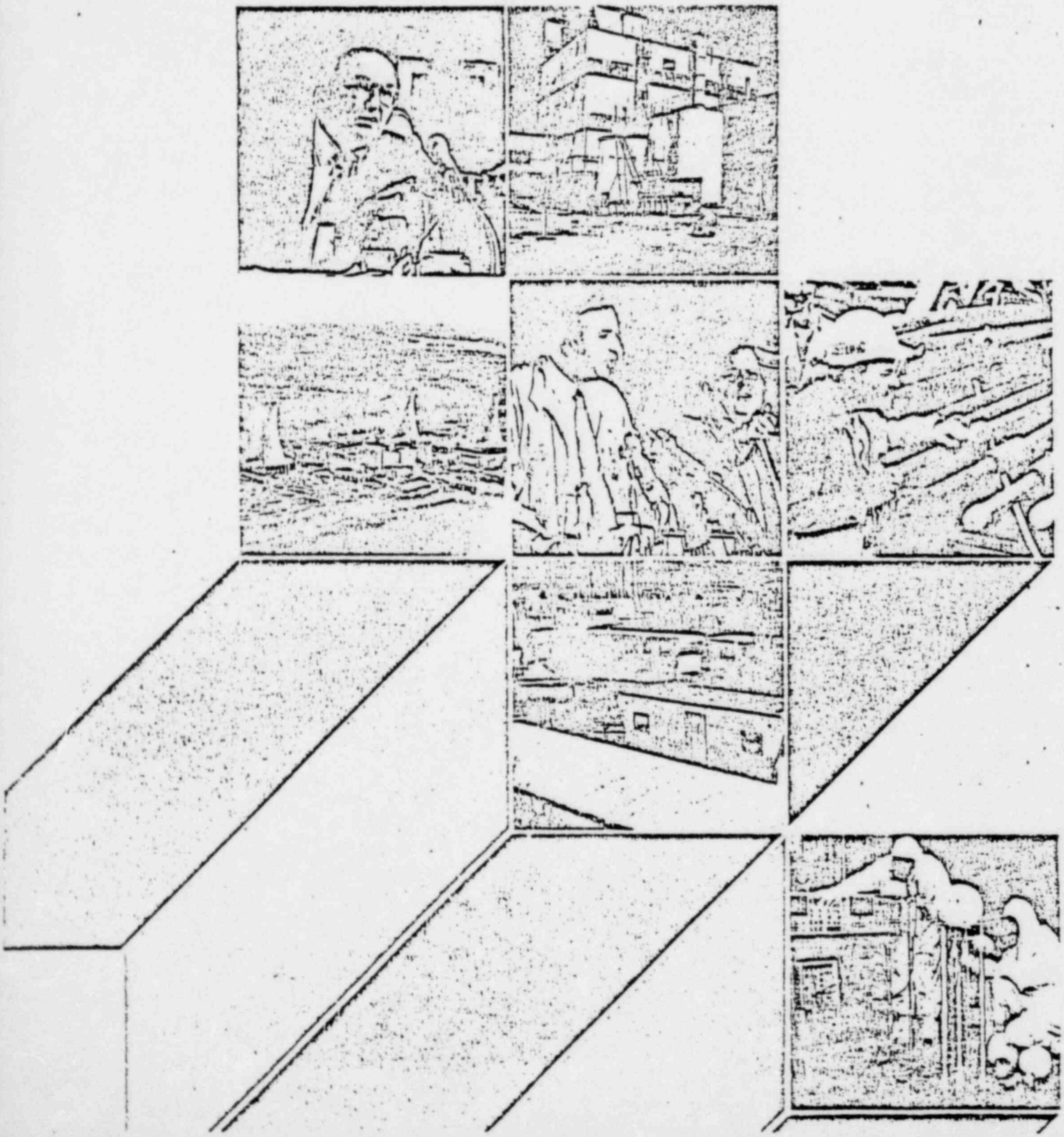
(3) LILCO should demonstrate that no furnace-sensitized material remains in the RCPB (via assembly of records, tests, etc. as needed).

(4) LILCO should provide verification that all reactor internal components constructed of 304 stainless material are in the solution heat treated condition and conform with the provisions stated in FSAR Section 4.5.

EXHIBIT 1

U.S. NUCLEAR
REGULATORY COMMISSION

1979
Annual Report



The President
The White House
Washington, D.C. 20500

Dear Mr. President:

Enclosed is the fifth Annual Report of the United States Nuclear Regulatory Commission for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974. This report covers the major activities of the NRC from October 1, 1978 through September 30, 1979 and briefly describes some additional actions through 1979 into 1980.

Respectfully,

John F. Ahearne

I, II, and III containment designs. The results of Generic Task A-39 will be an integral part of the final acceptability of these designs. The portions of this generic task related to the Mark I and Mark II containments are currently scheduled to be completed in March 1980.

Anticipated Transients Without Scram

Nuclear plants have safety and control systems to limit the consequences of abnormal operating conditions transients. Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. "Anticipated Operational Occurrences" or "Anticipated Transients" are defined (10 CFR Part 50, Appendix A) as "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe anticipated transient and the reactor shutdown system did not function as designed, then an "anticipated-transient-without-scram," or ATWS, would have occurred.

This issue has been discussed throughout the NRC and AEC and the nuclear industry for a number of years. Details on the safety significance of the issue and actions taken by NRC and industry prior to fiscal year 1979 in response to it may be found in the 1978 NRC Annual Report, pp. 27 and 28.

On the basis of discussions with senior NRC management, the Advisory Committee on Reactor Safeguards, and industry representatives, and the review of the Lewis Committee report on the Reactor Safety Study, the NRC staff in December 1978 proposed a combination of preventative and mitigative means of providing protection from ATWS events. In this supplement, the NRC staff proposed different types of plant modifications. The design alternatives which were proposed take into consideration the status of the plants—whether operating, under construction or nearly ready for operation—and questions of practicability, including the cost of such modifications.

In order to confirm that these alternatives provided the needed level of safety, the industry was required to provide the necessary confirmation analyses and the staff originally intended to make its recommendations to the Commission in the spring of 1979. The Three Mile Island Unit 2 accident affected these plans in several ways. First, both industry and NRC staff manpower were diverted from ATWS work; second, the Three Mile Island event scenario indicated that a number of aspects of the ATWS accident evaluation required reconsideration, especially for PWRs.

The shortage of available industry manpower delayed several of the required submittals of confirmation analyses. The result was a substantial slip in the projected completion date for the ATWS task. NRC staff manpower was partially restored in June 1979 and meetings were held with industry representatives in July and August of 1979 to discuss the impacts of the Three Mile Island Unit 2 accident on their respective ATWS evaluations. As of January 1, 1980, the NRC staff planned to propose an ATWS rule to the Commission by April 15, 1980, with a goal of issuing a final ATWS rule by December 1980.

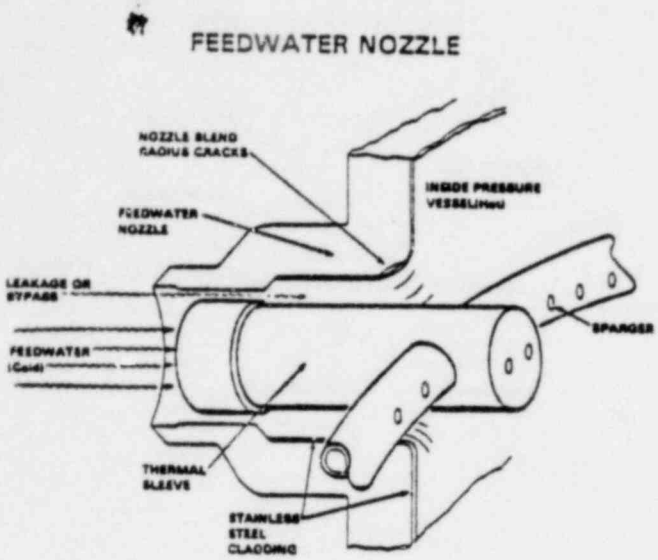
BWR Nozzle Cracking

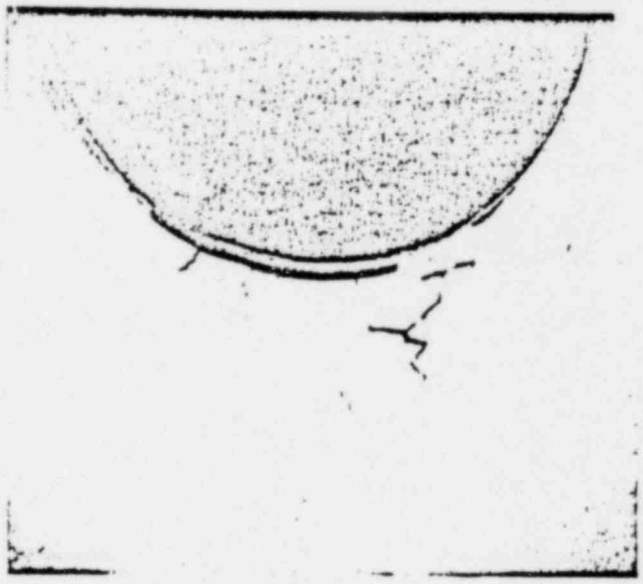
Over the last several years, inspections at 22 of the 31 boiling water reactor (BWR) plants licensed for operation in the U.S. have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at 18 facilities. One facility has not yet accumulated significant operating time and has, therefore, not yet been inspected.

The feedwater nozzles are an integral part of the primary pressure boundary of the reactor coolant system and the second barrier (after the fuel cladding) to the release of radioactive fission products. All of the repaired BWR feedwater nozzles met the ASME pressure vessel code limits, however, and no immediate action was necessary. Because only relatively small amounts of metal have been removed by repair operations, there has been no significant reduction in safety margins. Nevertheless, the cracking is potentially serious because:

- Excessive crack growth could lead to impairment of pressure vessel safety margins.
- The design safety margin could also be reduced by excessive removal of nozzle reinforcement while grinding out cracks, and repair by welding would be complicated.
- The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.
- The repair of these kinds of cracks can result in considerable shutdown time at the plant affected.

The reactor vendor (the General Electric Company) and the NRC have concluded from their separate studies that the cracks are initiated by rapid fluctuations in water temperature on the inside surface of the nozzles during periods of low feedwater temperature when flow may also be unsteady and perhaps intermittent. The cracks then grow deeper as a result of operational startup and shutdown cycles or other operationally induced transients. The stainless steel cladding exhibited less resistance to crack initiation than the underlying low-allow steel.





Cracks in nozzles of the feedwater and control rod drive lines of BWR reactor pressure vessel have been studied by the vendor (General Electric) and the NRC staff for several years. Evidence indicates that abrupt and wide fluctuations in water temperatures (see diagram above) are the initial causes of cracking. Photo above shows such cracks.

The vendor has performed extensive analysis and testing to confirm the suspected cause of the cracking and to develop possible long-term solutions—a newly designed sleeve, removal of the stainless steel cladding, reduction of the temperature differential at the nozzle, or some combination of these. The licensees involved have increased the number and extent of in-service inspections of feedwater nozzles, with careful repair and reinspection where cracks were found. The vendor advised these licensees to closely monitor startup and shutdown procedures in an effort to substantially reduce the time during which cold feedwater is being injected into the hot pressure vessel.

In a closely related area, the NRC was informed in March 1977 by the General Electric Company that a crack had been found in the nozzle of the control rod drive (CRD) return line in a reactor vessel. The CRD return line nozzles are the openings in BWR pressure vessels through which the high pressure water in excess of that needed to operate and cool the CRDs is returned to the pressure vessel. The cracks resembled those found in the feedwater nozzles and seemed to be the result of the same kind of cyclic thermal stresses that were causing feedwater nozzle cracks. The maximum crack depth has been 0.87 inch.

The NRC staff efforts related to the resolution of these two similar issues regarding nozzle cracking in boiling water reactors were consolidated into a single staff effort, Generic Task A-10, in 1977. Under Generic Task A-10, the staff issued interim guidance to operating plants in a report entitled, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," in 1977.

The staff has now completed its review of the General Electric studies on feedwater nozzle cracking and has concluded that the new sparger design—in conjunction with other remedial measures, such as clad removal and more appropriate operating procedures—is an effective means of greatly reducing the probability of crack initiation. The new sparger design includes flow discharge nozzles, a triple thermal sleeve, and two piston ring seals in the nozzle bore. The effectiveness of the new design in minimizing bypass leakage and other problems encountered in the older designs was confirmed by extensive testing and analyses by General Electric, including vibration, thermal-hydraulic, materials, and thermal fatigue evaluations. Other designs may also be acceptable.

Feedwater system changes, necessary on some low feedwater temperature plants to assure no cracking over the lifetime of the plant, are being evaluated on a plant-specific basis. An NRC staff report incorporating guidance for operating reactors and plants under licensing review is in preparation and is scheduled to be issued for comment in February 1980.

The resolution of questions regarding the future selection of improved inservice inspection techniques and frequency of inspection has been separated from the generic task while major industry investigations continue (including thermal cracking in a full-size nozzle mockup to be used in ultrasonic evaluation). A supplement to the NRC staff report cited above may be necessary upon completion of these studies. In the meantime, stringent inspection requirements, based mainly upon dye-penetrating testing, are still in force. All licensee efforts, such as system and operational

changes, to lengthen the time to crack initiation and to slow crack growth are taken into account in the determination of inspection techniques and criteria.

The CRD nozzle issue will be resolved by a combination of actions which includes nozzle inspection and repairs and some CRD system notifications. Certain system modifications recommended by General Electric involved cutting and capping the nozzle and return line but that action would reduce the capability to direct high pressure water through the CRD system when the vessel is otherwise isolated. Although this system is not normally expected to perform this function in safety analyses, the capability played a major role in keeping the core covered during the incident at Browns Ferry Unit 1 on March 22, 1975. As a result of its review of these modifications, the NRC has concluded that only a limited number of plants will be allowed to modify the CRD system in accordance with the GE recommendations. Unless the licensees of the remaining plants demonstrate, by testing, that sufficient flow is available to the reactor vessel with the return line removed, they will be required to retain the return line, rerouted to the feedwater line or a similar suitable connection that doesn't have the potential for cracking in the reactor vessel nozzle. The staff's evaluation, conclusions, and guidance on the CRD return line nozzle issue will also be included in the February 1980 NRC staff report referred to above.

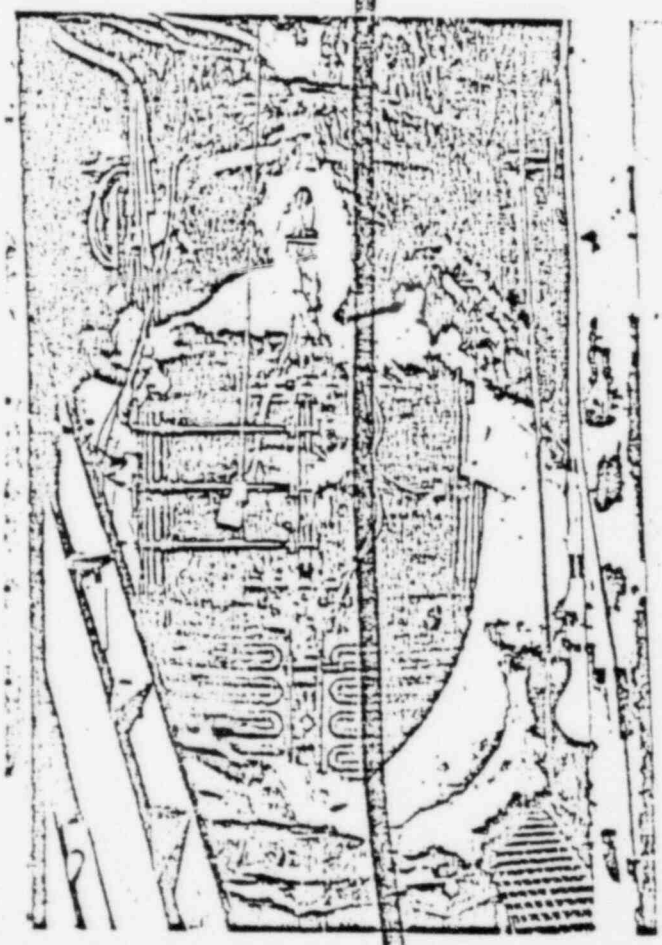
Plant-specific implementation of the generic licensing positions developed under this task (with the exception of future inservice inspection questions) has already begun.

Reactor Vessel Material Toughness

Nuclear reactor pressure vessels are required to have an adequate margin of protection against fracture in the presence of relatively large postulated flaws. This requirement is imposed for the sake of conservatism, even though extensive, periodic inservice inspection programs serve to provide protection against the presence of such flaws. Fracture mechanics—the engineering method used to establish the failure margin—employs a quantitative material property called fracture toughness to calculate the conditions under which catastrophically rapid crack propagation will occur. Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three facts are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, the technical specifications for power reactors set limits on the operating pressure during heatup and cooldown operations. These restrictions assure that the combination of pressure and temperature will remain well below that which might cause brittle fracture of the reactor vessel if a significant flaw were present in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these technical specifications over the life of the plant.

For the service time and operating conditions typical of current operating plants, reactor vessel fracture toughness provides adequate margins of safety against vessel failure. Further, for most plants the vessel material properties are such that adequate fracture toughness can be maintained over the life of the plants. However, results from a reactor vessel surveillance program indicate that up to 20 older operating pressurized water reactor pressure vessels were fabricated with materials that will have marginal toughness after comparatively short periods of operation. This issue has been incorporated in the



The protective insulation has been pulled aside following the testing of a weld-repair portion of a six-inch thick pressure vessel. A flaw more than five inches deep and 15 inches long was created in the area which was then subjected to pressure overloads more than double the design pressure, without disruptive failure.

for the movement of heavy loads over spent fuel to assure that the potential for a handling accident that could result in damage of spent fuel is minimized while the generic evaluation proceeds. In addition, the licensees were requested to provide information on load handling operations for use in the Task A-36 review. Responses were received from all licensees by December 1978.

The staff has completed its survey of load handling operations at operating plants, including design and procedural measures that prevent or mitigate the consequences of a heavy load handling accident and has prepared a draft report containing the NRC staff's resolution of this issue including revised criteria and other recommendations. This report is expected to be issued for public comment in January 1980. The report will provide the basis for revisions to the Standard Review Plan (SRP) and Regulatory Guides, if needed, that can be used in future reviews of new plants and will provide the basis for implementing additional requirements and procedures in operating plants.

Although Task A-36 will result in generic criteria, implementation of these criteria will be dependent on plant design characteristics and the specific procedures in effect at each particular plant, and will consequently require a plant-by-plant review.

Seismic Design Criteria

NRC regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants is provided in the NRC regulations and in Regulatory Guides. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken (principally as part of the Commission's Systematic Evaluation Program) to assure that these plants do not present an undue risk to the public.

The NRC staff is conducting Generic Task A-40 as part of the NRC Program for Resolution of Generic Issues. Task A-40 is a compendium of short-term efforts to support the reevaluation of the seismic design of operating reactors, and to support licensing activity in general. The objective of the task is, in part, to investigate selected areas of the seismic design sequence to determine the conservatism for all types of sites, to investigate alternative approaches to part of the design sequence, and to estimate quantitatively the overall conservatism of the design sequence. In this manner the program will aid the NRC staff in performing its reviews of the seismic design of operating reactors.

The NRC Office of Nuclear Regulatory Research is also undertaking a related but more comprehensive and long-term program to develop mathematical models to realistically predict the probability of radiactive releases from seismically induced events in nuclear power plants. This Seismic Safety Margin Research Program will utilize input from Task A-40 in a number of areas.

Generic Task A-40 is subdivided into two phases. Phase I includes a number of subtasks related to the response of structures, systems, and components to earthquakes. These subtasks include studies on: (1) quantifying conservatisms in seismic design, (2) electro-plastic seismic analysis methods, (3) site-specific response spectra, (4) nonlinear structural dynamic analysis procedures, and (5) soil structure interaction. These studies were performed under NRC-sponsored contracts and all were completed by October 1979. Review of the results of these studies is underway. The results will support the effort on seismic reevaluation of operating plants, particularly in the area of site-specific definition of seismic input. As of January 1, 1980, Phase I was scheduled to be completed in February 1980, with the issuance of recommendations for changes in the Standard Review Plan and Regulatory Guides in those seismic design areas related to response of structures, systems, and components to seismically induced events.

Phase II of Task A-40 includes several subtasks related to numerical modeling of earthquake motion at the source, analysis of near source ground motion, and attenuation of high frequency ground motion. Studies under these subtasks being conducted by NRC contractors are scheduled for completion by the end of 1980. Review and implementation of the results of these studies in terms of recommended revisions to the Standard Review Plan and Regulatory Guides are scheduled for March 1981.

Pipe Cracks at Boiling Water Reactors

Pipe cracking has occurred in the heat affected zones of welds in primary system piping in BWRs since the mid-1960s. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure mode by being "sensitized," either by welding or by post-weld heat treatment. Although the likelihood is extremely low that IGSCC-induced cracks will propagate far enough to create a significant hazard to the public, the occurrence of such cracks is undesirable and measures to minimize IGSCC in BWR piping systems are indicated to improve overall plant reliability.

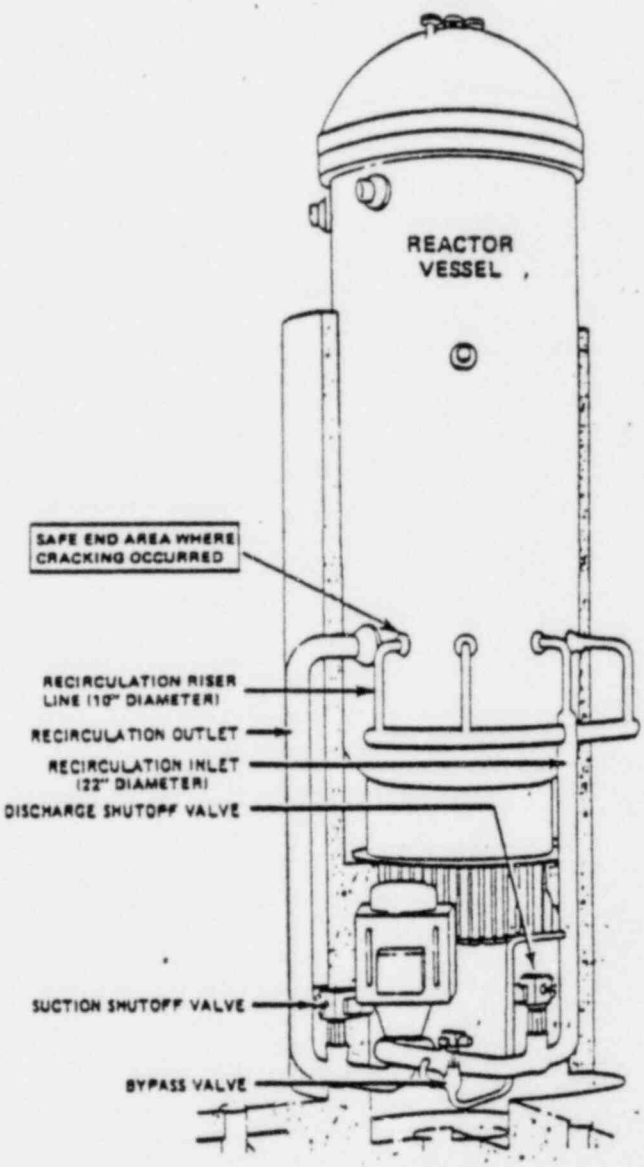
"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were found to be susceptible to IGSCC in the late 1960s. Because they were susceptible to cracking, the Atomic Energy Commission took the position in 1969 that furnace-sensitized safe ends in older plants should be removed or clad with a protective material, and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, moreover, are in small diameter lines and are subjected to augmented inservice inspection.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop bypass lines and in 10-inch diameter core spray lines. More recently cracks were discovered in recirculation riser piping (12- to 14-inch) in all foreign plants. All these crack locations are part of the reactor primary system. Cracking is most often detected during inservice inspection using ultrasonic testing techniques. Some piping cracks have been discovered as a result of small primary coolant leaks.

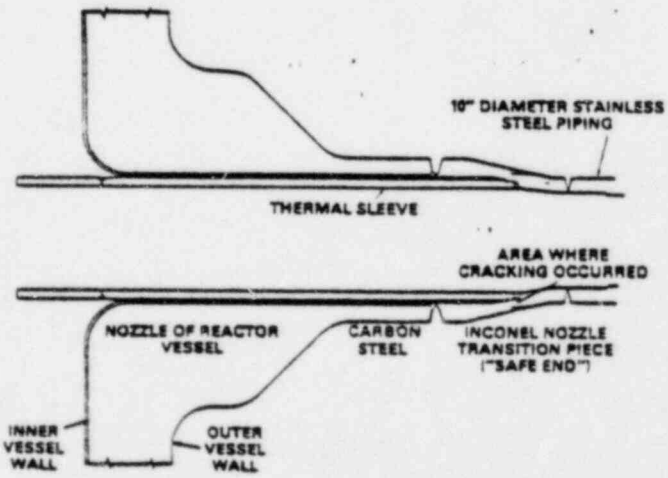
In response to these occurrences of BWR primary system cracking, a number of remedial actions were undertaken by the NRC. These actions included:

- Issuance of Regulatory Guide 1.44 on "Control of the Use of Sensitized Stainless Steel."
- Issuance of Regulatory Guide 1.45 on "Reactor Coolant Pressure Boundary Leakage Detection Systems."
- Closely following the incidence of cracking in BWRs, including foreign experience.
- Encouraging replacement of furnace sensitized safe ends.
- Requiring augmented inservice inspection of lines having less corrosion resistant stainless steel, especially those that have a high potential for cracking (service sensitive lines).
- Requiring upgrading of leakage detection systems.

More recently pipe cracking and furnace-sensitized safe end cracking have been reported in larger (24-inch diameter) lines in a GE-designed BWR in Germany with over 10 years of service. Because the safe ends in that facility had been furnace-sensitized during fabrication, IGSCC was suspected. As a result of concerns regarding these furnace-sensitized safe ends, a safe end was removed and subjected to destructive examination. During laboratory examination of the removed safe end, including a small section of attached pipe, cracks were discovered at various locations in the safe end and in the weld heat affected zone of the pipe. The cracks in the pipe weld area were very shallow with the maximum depth less than 5 mm (about 1/8-inch) in a wall thickness of about 1.5 inches. Cracking in the furnace-sensitized safe end, also having a wall thickness of about 1.5 inches, was



Cracks appeared at one facility in the so-called "safe ends" between pipes and vessel nozzles. This diagram shows the general location of the safe-end area (above), and detail of the cracking area (below).



somewhat deeper. The German experience was the first known occurrence of IGSCC in pipes as large as 24 inches in diameter.

In June 1978, a through-wall crack was discovered in an Inconel recirculation riser safe end (10-inch diameter) at the Duane Arnold facility. The crack has been attributed to IGSCC, although the material in this instance is different from the Type 304 stainless steel that has been historically found to be susceptible to IGSCC. Prior to safe end removal, ultrasonic examination showed several indications of possible cracks. Following their removal, cracking was discovered in all eight safe ends. The cracking appeared to have originated in a tight crevice between the inside wall of the safe end and the internal thermal sleeve attachment. Such crevices are know to enhance IGSCC. Differences in materials, geometry, stress levels, and crevices appear to make the problem at Duane Arnold unique to a particular type of recirculation riser safe end (Type I). As a result of this event, ultrasonic examination of the other Type I safe ends in U.S. BWRs (i.e., at the Brunswick 1 and 2 facility) was conducted. No significant indications of possible cracks were found in Unit 2 and one indication was identified at Unit 1. Although this latter indication was relatively minor and too small to be reportable pursuant to the NRC Regulations, periodic reevaluation of the Unit was deemed necessary. This ultrasonic indication at Brunswick Unit 1 was remeasured and reevaluated in the presence of NRC ultrasonic testing consultants at another plant shutdown in January 1979. It was concluded that: (1) there is no apparent change of this indication between inspections, and (2) although the existence of a very small localized area of cracking cannot positively be ruled out, the most likely cause of this indication is irregularities at the weld-to-base metal interface of the first bead weld at the thermal sleeve to safe end weld. This indication will be reexamined during the next refueling outage.

General Electric (the reactor vendor) has been asked to provide an in-depth report on the significance of recent events, including current inspection, repair, and replacement programs. They were also asked to address any new safety concerns related to the occurrence of cracking in large main recirculation piping. Based on information presented by General Electric to date and on extensive staff evaluation, the staff concluded that the recent occurrences do not constitute a basis for immediate concern about plant safety, nor require any new immediate actions by licensees.

Based on the earlier incidents of pipe cracking discussed above, the NRC formed a Pipe Crack Study Group to: (a) investigate the cause of cracks, (b) make interim recommendations for operating plants, and (c) recommend corrective actions to be taken for future plants. The Study Group published its report (NUREG-75/067) in October 1975, containing recommendations to reduce the incidence of IGSCC in sensitized stainless steel piping. Following staff review of the Study Group's recommendations, the staff issued an implementation document (NUREG-0313) which established staff positions consistent with the recommendations of the Study Group.

As a result of the more recent incidents, the NRC reestablished a second Pipe Crack Study Group on September 14, 1978. The new Study Group specifically addressed the following issues:

- The significance of the cracks discovered in large diameter pipes relative to the conclusions and recommendations set forth in the referenced report and its implementation document, NUREG-0313.
- Resolution of concerns raised over the ability of ultrasonic techniques to detect cracks in austenitic stainless steel.
- The significance of the cracks found in large diameter sensitized safe ends, and any recommendations regarding the current NRC program for dealing with this matter.
- The potential for stress corrosion cracking in PWRs.
- The significance of the safe end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The new Study Group completed its evaluation in February 1979 and issued a report, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" (NUREG-0531). The new Study Group not only reaffirmed the conclusions and recommendations reached by the previous group in NUREG-75/067, but also presented some new ideas to reduce the potential for IGSCC and addressed the subject of IGSCC in safe ends. On March 13, 1979, NRC issued a Notice in the Federal Register soliciting public comments on NUREG-0531. After expiration of the public comment period and review of the Study Group's conclusions and recommendations, the staff initiated Task A-42. The work to be performed under Task A-42 was defined at that time as the development of an update to the implementation document, NUREG-0313, to incorporate the new Study Group's conclusions and recommendations and public comments received on NUREG-0531.

Revision 1 to NUREG-0313 was issued in October 1979, and public comments have been solicited on the report. Revision 1 sets forth the NRC staff's revised guidelines for reducing the IGSCC susceptibility of BWR piping. The guidelines describe a number of preventive and corrective measures acceptable to the NRC, including guidelines for: (1) corrosion resistant materials for installation in BWR piping, (2) methods

of testing, (3) processing techniques, (4) augmented in-service inspection, and (5) leak detection. The report also included recommendations for developmental work to provide future improvements in limiting the extent of IGSCC or detecting it when it occurs.

Containment Emergency Sump Reliability

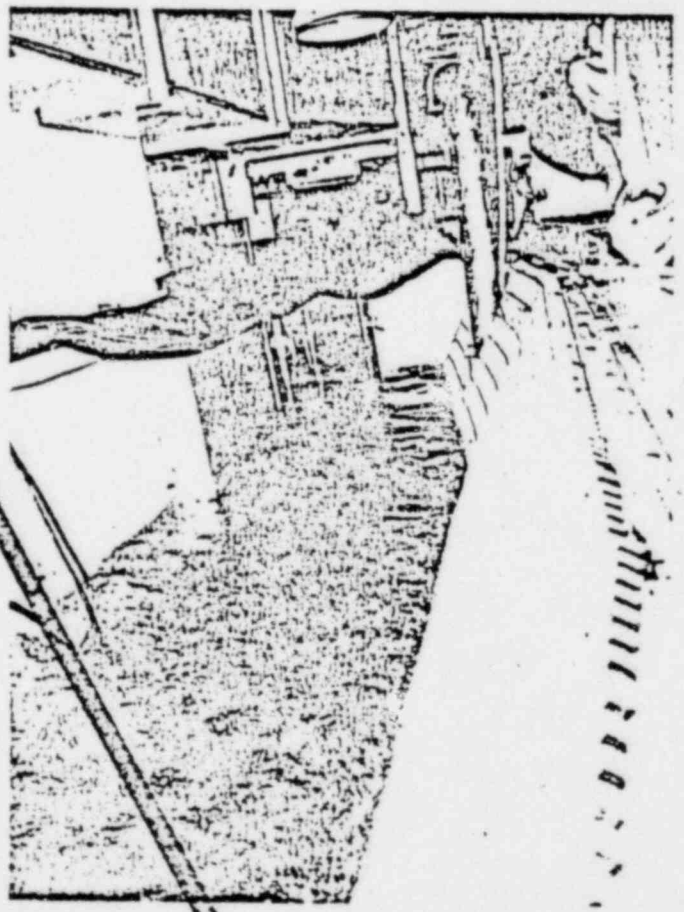
Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the emergency sump at the low point in the containment. This water would later be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could therefore disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or loss of integrity in the containment.

One potential way the ability to draw water from the emergency sump can be lost is from blockage by debris. A principal source of such debris could be the thermal insulation normally installed on the reactor coolant system piping. In the event of a piping break, the subsequent violent release of the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. The loose insulation material could then be swept into the sump and block it.

A Task Action Plan was under development in March 1979 when the Three Mile Island Unit 2 accident disrupted work on it. As of January 1, 1980, the Task Action Plan was nearing completion. Nonetheless, several technical studies related to sump reliability which were already underway will either be incorporated into Task A-43 or will provide input into Task A-43 efforts.

A study program investigating PWR vortex technology has been completed by the Iowa Institute of Hydraulic Research and a technical report issued. A summary report of NRC experience with containment sump testing is being prepared. This summary will be issued as a NUREG report in 1980. Based on the Iowa study program and the review of tests, NRR expects to draft interim positions on sump design guidelines and preoperational test requirements in early 1980. Criteria for the evaluation of operating containment sumps will be formulated at about the same time.

Finally, a program is being sponsored by the Department of Energy, in cooperation with NRC, to aid in resolving this issue as part of their Light Water Safety Research Program. This is an experimental pro-



NRC staff members traveled to the North Anna Power Station Unit 1 in Virginia to conduct evaluations of the emergency recirculation sump as part of its work on Generic Issues Task A-43. The photo at the top shows project personnel looking down into the recirculation sump area. Above, the reactor containment area was deliberately flooded to permit observation of flow patterns, blockage of pipes, types of debris, etc.

EXHIBIT 2

LILCO
LONG ISLAND LIGHTING

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

May 15, 1981

FOR ENCLOSURES TO DOCUMENTS, REQUEST BEHIND THE FILE MATERIAL

SNRC-566

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

Forwarded herewith are six (6) copies of LILCO's responses to the Safety Evaluation Report (SER) Outstanding Issues listed in Attachment 1. Responses to the remaining SER Outstanding Issues, listed in Attachment 2 are scheduled to be submitted on May 27, 1981, unless otherwise noted.

Please note that our response to Outstanding Issue Number 8 "Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment" has been forwarded to you under separate cover via letter SNRC-561, dated May 15, 1981.

Very truly yours,

J. P. Novarro

J. P. Novarro
Project Manager
Shoreham Nuclear Power Station

RWG:mp

Enclosures

cc: J. Higgins

LTR B001 S 1/6
ENCL B028 S 1/6

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ATTACHMENT 1 - SER OUTSTANDING ISSUES

Number	Issue
3	Piping Vibration Test Program - Small Bore Piping/Instrument Lines
5	LOCA Loadings on Reactor Vessel Supports and Internals
13	NUREG-0619, Feedwater Nozzle and Control Rod Return Line Cracking
14	Jet Pump Hold-down Beam
18	NUREG-0313, Rev. 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping
22 *	Appendix G - Impact Testing
36	Containment Purge System
38 *	Fracture Prevention of Containment Pressure Boundary
44	Level Measurement Errors
46	OIE Bulletin 79-27
50	Low and/or Degraded Grid Voltage Condition
51 *	Fracture Toughness of Steam Line and Feedwater Line Materials
56	Financial Qualifications (Supplement to SNRC 561 dated April 30, 1981, issue 56)
61	Scram Discharge Volume

* BOP portion of response only

SER OPEN ITEM #13

NUREG-0619
"FEEDWATER NOZZLE AND
CONTROL ROD RETURN LINE CRACKING"

RESPONSE:

Control Rod Drive REturn Line Nozzle Cracking

Postmodification CRD-System Performance Test

The CRD System will be tested to prove satisfactory system operation; return flow capability will be demonstrated to be equal to or in excess of the flow required to satisfy the base-case conditions and two-CRD-pump operation, if concurrent operation of two CRD pumps are required to achieve this flow.

CRD Hydraulic System Maintenance Procedures

For Shoreham, all lines are constructed of stainless steel and do not require procedural modifications for flushing the exhaust-water header and cleaning the filters in the insert and exhaust lines.

Triple Sleeve Sparger Design

The welded thermal sleeve feedwater sparger has been replaced with an improved interference fit sparger,also called the triple sleeve sparger, for the Shoreham Nuclear Power Station.

Clad Removal

Feedwater nozzle cladding will not be used for the Shoreham Nuclear Power Station.

System Modifications

(a) Low Flow Feedwater Controller - the Shoreham Nuclear Power Station low flow feedwater control system consists of two low flow control valves in parallel with split range control. The split range control enables control to below 0.1% of rated flow. The basic purpose of utilizing a dual element control scheme is to limit feedwater temperature fluctuations to within 50°F peak-to-peak on a continuous basis. Feedwater temperature variations will be monitored during start-up to evaluate the need for additional controls. If necessary, administrative controls and/or procedures will be developed to guide the operators during low flow operation to further minimize thermal cycling.

(b) Reactor Water Cleanup System (RWCU) - RWCU discharge currently has the capability to deliver flow to all feedwater nozzles at Shoreham.

Operating Procedure Modifications

The following procedural modifications will be evaluated for Shoreham Plant operation:

(a) RWCU flow will be directed to all feedwater nozzles at maximum flow rate and exit temperature during all low flow conditions prior to turbine loading.

(b) Low load, reduced pressure synchronization of the turbine will be evaluated during start-up.

(c) Maximum feedwater heating will be initiated as soon as possible by isolating all feedwater heaters except the highest stage of feedwater heating.

(d) During start-ups and shutdowns, the feedwater control system will be operated to maintain low flow control sufficient to eliminate on-off feedwater operation and with sufficient controllability to preclude greater than 25°F peak-to-peak mixture temperature variations during steady demand.

(e) In general, plant operating procedures will be modified to minimize subcooling, particularly at high feedwater flow rates.

Pre-Service Inspection

(a) Performance of PT has been accomplished for each nozzle prior to the installation of the sparger.

(b) Performance of a baseline UT will be accomplished for each nozzle after the installation of the sparger. Test results will be retained with the plant's permanent records.

In-Service Inspection

(a) At every second scheduled refueling outage, an external UT will be performed of all feedwater nozzle safe ends, bores and inside blend radii. If nozzle cracks are indicated, the sparger will be removed and a PT of the nozzle bore and blend radii will be performed; and repairs will be made as required.

(b) At every fourth scheduled refueling outage, a visual inspection of all spargers will be conducted.

(c) At every ninth scheduled refueling outage (or after 135 startup/shutdown cycles) a PT examination will be performed. The PT will include removal of a sparger from one nozzle followed by flapper wheel grinding and PT examination of both the nozzle of the removed sparger and the accessible portions of the other nozzles.

Shoreham Outstanding SER Issue #18

NUREG 0313

"Technical Report on Material Selection" and Processing Guidelines for BWR Coolant Pressure Boundary Piping

Both NRC (Pipe Cracking Study Group) and industrial study groups such as GE and EPRI "Owner's Groups" have been investigating the phenomena of BWR stress corrosion cracking for a number of years. One common conclusion that has been drawn from these studies is that the stress corrosion cracking issue is not a safety issue. The conclusion that austenitic stainless steel will "leak before breaking" is substantiated and supported by all the above groups. In addition, no incident to date has led to a major release of radioactivity to the environment, and the relatively small cracks have been identified either by leak detection or ISI. Therefore, we believe the stress corrosion cracking issue is one of reliability, not safety, and should be addressed accordingly. Although the Shoreham plant's initial design did contain many of the "service sensitive" lines outlined in NUREG 0313, Shoreham has been cognizant of these industry problems and has taken steps to mitigate future concerns about stress corrosion cracking.

Accordingly, we believe that the problem of stress corrosion cracking has, to a large degree, been eliminated for Shoreham. In particular: (a) the recirculation bypass line has been eliminated, (b) the core spray line and safe end materials have been changed and, (c) the CRD return line has been eliminated. Therefore, the only primary pressure boundary piping which has remained 304 stainless steel is the reactor recirculation system. A modification of the piping material for this system to an alternative material was not practical in the time table for Shoreham.

In an effort to further reduce the potential for stress corrosion cracking in the recirculation system, sections of the shop fabricated piping were solution heat treated prior to field erection. These piping sections were received from GE in a non-heat treated condition. The heat treating should eliminate all concerns about stress corrosion cracking for the shop welds that were heat treated. For the field welds, LILCO did institute a number of measures whereby the welding techniques used were modified such that sensitization was minimized and the welding residual stresses introduced during the welding process were reduced to the best extent practical. These measures included: (1) welding process control (i.e., preheat temperature and interpass temperature control), (2) grinding restrictions, and (3) weld filler material ferrite control as required by Regulatory Guide 1.33.

NUREG-0313, Revision 1

To address NUREG-0313 specifically for Shoreham, the ASME Code Class 1 and Class 2 reactor cooling pressure boundary piping meets the guidelines stated in Part 2 except for portions of the reactor recirculation system (B31) as described above, and, stainless steel to carbon steel transition welds between the Recirculation System and the Reactor Water Cleanup, Core Spray and Residual Heat Removal System. As stated previously, only field welds and shop welds classified as "nonconforming" would fall under the requirements for Part 3 of NUREG-0313.

Part 3:

A. 1.0 ISI

The recirculation system welds and transition welds classified as "nonconforming" will be inspected in accordance with Part III B1.0 of NUREG-0313 to the extent practicable for Shoreham Nuclear Power Station. Attempts will be made to inspect all welds with the appropriate ISI UT techniques, but physical interference in some locations may preclude the inspection. Although NUREG-0313 Revision 1 implies that recirculation riser lines and recirculation inlet lines at safe end curves should be considered "Service Sensitive", we would argue that this has not been documented for BWR 4 designs such as Shoreham. First, most of the riser sections for Shoreham have been solution heat treated and, therefore, IGSCC is not a concern. For the recirculation system heater to riser welds these welds will be inspected as Nonconforming Lines, but non-Service Sensitive. For recirculation nozzles/thermal sleeve designs similar to Shoreham (i.e., Fitzpatrick), no evidence of cracking has been discovered. Therefore, until further IGSCC incidents are documented in these areas we believe it is appropriate to classify these areas as Nonconforming, but Non-Service Sensitive.

2.0 Leak Detection

The Shoreham primary containment leakage detection system is in full compliance with Regulatory Guide 1.45. The technical specification limits for reactor coolant leakage will be includes as part of the Technical Specification submittal for Shoreham. We believe a change in limiting conditions for leakage is not warranted since this issue is not a safety concern and thus the present limiting conditions are acceptable.

If any cracks are detected, all spargers will be removed and a complete examination of all nozzles will be conducted. All nozzle cracks will be evaluated and repaired as required.

(d) An on-line monitoring system for detecting leakage through degraded seals and/or cracks in thermal sleeve welds will be evaluated and considered for Shoreham feedwater nozzle application after fuel load, in lieu of continued in-service inspection.

EXHIBIT 3

5.2.3 General Material Considerations

5.2.3.1 Material Specifications

Table 5.2.3-1 lists the principal pressure retaining materials and the appropriate material specifications for the RCPB components.

5.2.3.2 Compatibility with Reactor Coolant

The materials of construction exposed to the reactor coolant consist of the following:

1. Solution annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316, and 316L.
2. Nickel base alloys - Inconel 600 and Inconel X750.
3. Carbon steel and low alloy steel.
4. Some 400 series martensitic stainless steel (all tempered at a minimum of 1100 F).
5. Colmonoy and Stellite hardfacing materials.

All of these materials of construction are resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon or low alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Radiolytic products in a BWR have no adverse effects on the construction materials.

5.2.3.2.1 Use of Sensitized Stainless Steel

The use of severely sensitized or furnace sensitized stainless steel is prohibited. However, exception is taken to Regulatory Guide 1.44 which effectively prohibits the use of welded type 304 stainless steel by considering it the same as severely sensitized material and by placing limits on oxygen concentration which cannot be met by operating BWRs.

The stated objective of the Regulatory Guide is to control the application and processing of stainless steel to avoid severe sensitization "to diminish the numerous occurrences of stress corrosion cracking in sensitized stainless steel components of nuclear reactors."

Welded 304 (and 316) stainless steel is placed in the same category by the guide as severely sensitized stainless steel. An intergranular corrosion test is required for each welding procedure to be used for welding of material with carbon content greater than 0.03 percent. Limits are imposed upon water quality which are inconsistent with Regulatory Guides 1.37 and 1.56. The use of welded 304 stainless steel is prohibited, primarily by imposing an oxygen limit of 0.10 ppm which is not obtainable in the BWR during normal operation.

Our position is that present controls administered by General Electric Company (GE) for processing and application of welded 304 stainless steel are entirely adequate, have been well proven by excellent operating experience, and therefore the requirements imposed by the Regulatory Guide in its present form are not warranted.

That there have been instances of stress corrosion cracking (SCC) in welded 304 stainless steel in some early GE BWRs is indisputable. Also, there have been instances of SCC in furnace sensitized stainless steel. However, there have been no instances of SCC of 304 stainless steels which were processed with presently implemented controls.

5.2.3.2.2 Process Controls for 304 Stainless Steel

Improvements in technology fostered by extensive GE-APED research and development since the early GE BWRs 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.)

EXHIBIT 4

SECTION 6.1 - 6.3 of NUREG-0531

6.0 PIPE CONFIGURATIONS AND STRESSES

6.1 INTRODUCTION

There are approximately 17,000 (6.1) welds[a] in stainless-steel piping in operating BWRs. There have been 133 IGSCC incidents[b] in BWR stainless-steel piping systems. The question arises: Why have less than 1% of the welds cracked while more than 99% of the welds are still performing satisfactorily? The variations in detailed material characteristics and environmental conditions are probably major contributors to the 1% failure. However, the variability of stresses at welds may also help to explain why less than 1% of the welds have cracked.

Stresses at welds are caused by fabrication processes (e.g., welding) and by operation loads (e.g., internal pressure). Stresses from both fabrication processes and operation loads are dependent upon the pipe configuration. Accordingly, this chapter will outline first significant aspects of pipe configurations and then discuss fabrication and operating stresses and how those stresses can be influenced by configurations.

6.2 PIPE CONFIGURATIONS

6.2.1 Piping Systems

Table 6.1 lists typical BWR piping systems that are involved in the operation and/or safe shutdown of the reactor. Figure 6.1 shows the basic configuration of BWR recirculation

TABLE 6.1. List of Typical Piping Systems Involved in the Operation and/or Safe Shutdown of BWRs[a]

Identification Number	Function
1	Reactor Recirculation
2	Main Steam
3	Feedwater
4	Reactor Water Cleanup
5	Reactor Core Isolation Cooling
6	Core Spray
7	Residual Heat Removal
8	Containment Spray
9	Reactor Head Spray
10	Standby Liquid Control
11	High-Pressure Coolant Injection
12	Low-Pressure Coolant Injection

[a] This list is taken from the FSAR of Peach Bottom Units 2 and 3.

(a) The term "welds" is used in this chapter to refer to girth-butt welds.

(b) This does not include IGSCC that occurred in furnace-sensitized stainless-steel safe ends.

piping systems. While it does not show the various connections (e.g., residual heat removal) to the system or the support/restraining devices, it is sufficient for the present purpose because it shows that the recirculation piping system is a complex assembly of piping products consisting of several sizes of straight pipe, curved pipe and branch connections. Figures 6.2 and 6.3 are additional illustrations of the complex nature of BWR piping systems.

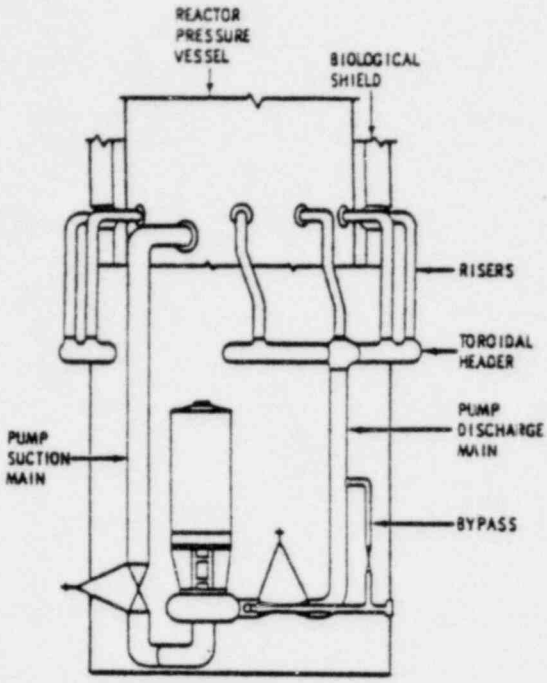

FIGURE 6.1. Basic Configuration of BWR Recirculation System
[Source: (6.2), p. 2.2]

6.2.2 Welds

Weld configurations are of particular significance to stresses in IGSCC. Pipe, fittings and valves are not precision-made products; the relatively large tolerances on diameters and wall thickness cause problems in achieving an adequate fit between two pipes (or pipe-to-fitting or pipe-to-valve) for butt welding. It is common practice to counterbore ends of pipe, fittings and valves to a standard "C-dimension," as illustrated in Figure 6.4. The locally-reduced wall thickness in the weld region may produce a significant increase in operation stresses as compared to such stresses remote from the weld region. The counterbore depth (d in Figure 6.4) is not a controlled dimension and usually varies with azimuth (around the circumference) location of a given pipe end. This may introduce variations in residual welding stresses with azimuth location.

There are many welds in piping systems. As typical examples, the Browns Ferry 1 recirculation piping system contains thirty 4-in., sixty 12-in., nine 22-in. and thirty-three 28-in. welds: a total of 132 welds. The Browns Ferry 1 core spray piping systems, up to the first isolation valves, contain six 10-in. and forty-eight 12-in. welds.

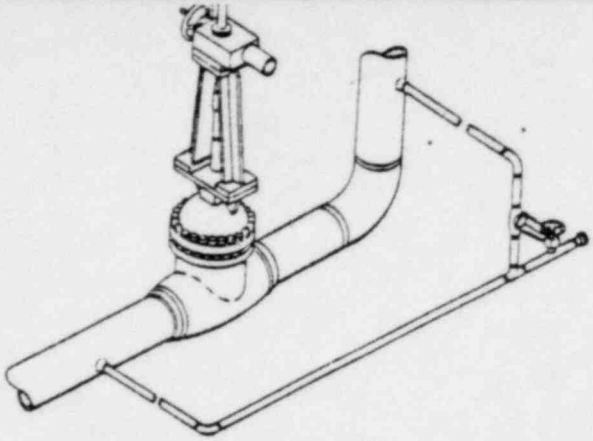

FIGURE 6.2. Dresden-2 Recirculation Bypass Lines - Loops A and B [Source: (6.2), p. 2.8]

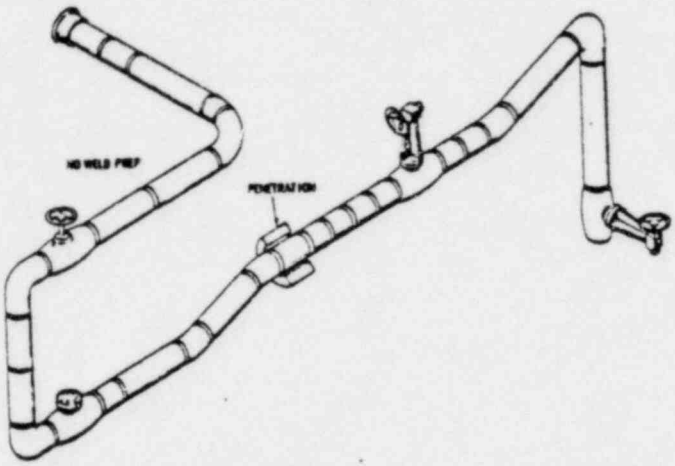

FIGURE 6.3. Dresden-2 Core Spray - Loop A (North) [Source: (6.2), p. 2.3]

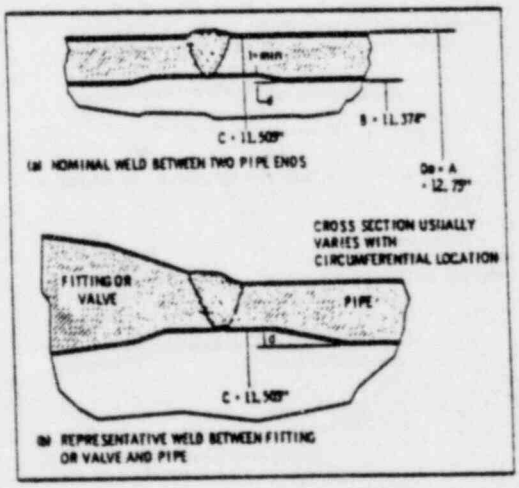

FIGURE 6.4. Configuration of Welds (Girth Butt Welds)

6.2.3 Other Detailed Configurations

Piping systems contain curved pipe, branch connections (including crosses in the recirculation loop, Figure 6.1) and other structurally complex shapes. Stresses in these products are different from those in straight pipe with the same operation loads. Further, welding residual stresses at welds between pipe and a valve, for example, may be different from stresses in a pipe-to-pipe weld. (As discussed in 6.3, some data are available on residual stresses in pipe-to-pipe welds.)

The Duane Arnold (see Chapter 7) recirculation inlet nozzle represents a type of detailed configuration of particular interest to this report. This type of configuration, in addition to the crevice problem, may involve relatively high stresses that are due to operation loadings such as pressure, pressure differential across the interior pipe, loads (possibly including vibration) transmitted to the attachment weld by the interior piping and differential thermal expansion between the interior pipe and the safe end or vessel nozzle. This type of detailed configuration is also encountered at the connection of interior piping to the core spray nozzles of the reactor vessel. The attachment weld will create residual stress and, if the pressure boundary material is susceptible, sensitization may occur.

6.3 FABRICATION STRESSES

Stresses (often called residual stresses) can arise because of forming processes (e.g., bending of pipe), welding, rough grinding or machining and forces required to obtain alignment for making the final weld (field closure weld) that unite the piping subassemblies into a piping system.[a] If the piping is annealed after these fabrication processes, the fabrication stresses are essentially eliminated. However, annealing after the field closure weld is usually not practical.

The most significant fabrication stresses in connection with IGSCC are probably due to welding and to rough grinding without annealing.

6.3.1 Weld Residual Stresses

Welding affects IGSCC in two ways. First, and perhaps most important, the high temperatures that occur during welding may sensitize the material adjacent to the weld. Second, welding produces residual stresses in the weld region; these residual stresses, either by themselves or in combination with operation stresses, provide the stress ingredient of IGSCC.

Test data (6.3, 6.4) are available on residual stresses at welds 4-in., 10-in. and 26-in. Schedule 80 Type 304 stainless-steel pipe-to-pipe welds. Test welds were made using procedures (e.g., number of passes, heat input) that were representative of standard nuclear-industry practices. Shack, Ellingston and Pahis include some residual stress data on welds in piping that was removed from BWR service (6.4).

The test data indicate that residual stresses in the region of welds vary in complex ways in all spatial directions; i.e., around the circumference (azimuthal location), along the axis of the pipe as a function of distance from the weld, and as a function of location

(a) In some piping systems, intentional forces at the field closure weld introduce "cold spring." In the BWR piping systems considered here, this is not done.

through the wall of the pipe. From the standpoint of IGSCC, the most significant residual stress is probably the tensile axial-direction stress on the inside surface within an axial distance of about 1/4 in. from the weld. The reasons for this are:

- Tensile stresses are more important than compressive stresses in IGSCC.
- Axial-direction stresses will cause cracks which follow along the edge of the weld and stay in the sensitized metal zone, in contrast to hoop stresses, which will cause axial cracks that tend to stop at the weld or upon leaving the sensitized metal zone.
- Cracks are most apt to begin on the inside surface.
- The 1/4-in. axial zone on each side of the weld appears to be the most severely weld-sensitized.

Giannuzzi (6.3) conducted tests on one 4-in., one 10-in. and two 26-in. Schedule 80 welds and found that the maximum tensile, axial-direction stress on the inside surface heat-affected zone (sensitized zone) could be ranked as follows:

highest:	4-in. Sch. 80
intermediate:	10-in. Sch. 80
lowest:	26-in. Sch. 80.

The stress in the 4-in. weld was about 20 to 25 KSI higher than in the 26-in. weld.

Shack, Ellingston and Pahis (6.4) conducted tests on three 4-in., one 10-in., one 26-in., Schedule 80 welds and found that the maximum tensile, axial-direction stress on the inside surface in the region 2 to 3 mm from the weld fusion line were:

4-in. welds:	38, 46, and 51 KSI for the three tests
10-in. weld:	60 KSI
26-in. weld:	28 KSI

The 10-in. weld was removed from a piping system in a BWR.

These conclusions suggest that welding residual stresses in 26-in. Schedule 80 pipe are less than in 4-in. Schedule 80 pipe. This is perhaps related more to the wall thickness of the pipe and number of passes (7 passes for 4-in., 30 passes for 26-in.) than to the size or diameter of the pipe.

Residual stresses are quite high, even in the 26-in. pipe tests. However, the differences between residual stresses in 4-in. and 26-in. pipe may be part of the reason why IGSCC is more frequent in small pipe than in large pipe.

Discussions of available test data (6.3, 6.4) lack rational explanations for the complex distributions and magnitudes of measured stresses. Efforts have been made to calculate residual stresses using finite element modeling of the structure and materials characteristics (6.5). However, at this time, the quantitative generality of conclusions drawn from the relatively scanty data is questionable. Also, tests were run on pipe-to-pipe welds,

whereas many welds in piping systems are pipe-to-fitting[a] or pipe-to-valve welds. It is not apparent that residual stresses in such welds will be equivalent to residual stresses in pipe-to-pipe welds.

6.3.2 Rough Grinding or Machining Stresses

Preparation of welding ends involve machining of a counterbore to obtain good alignment of the inside diameters at the pipe-to-pipe juncture; this alignment is needed to produce a weld with a smooth root and a weld that can be inspected. The counterboring is usually done by relatively rough machining; this can lead to high residual stresses on the counterbore surface. However, according to Shack, Ellingston and Pahis (6.4), welding tends to anneal-out these surface stresses in the weld-sensitized region close to the weld; hence machining or grinding before welding is probably not a major contributor to IGSCC. However, studies (6.3, 6.4) indicate that rough grinding after welding (as might be needed to smooth out an irregular weld root contour) can lead to high (up to 100 KSI) surface residual stresses. While these high, surface residual stresses extend only a few mils into the wall, they may initiate IGSCC.

6.4 OPERATION STRESSES

Operation stresses are those caused by operation loads such as internal pressure; weight of pipe, contained fluid and insulation; restraint of overall thermal expansion of the piping system; temperature gradients through the wall; and dynamic loads such as safety relief valve thrust, earthquakes and vibration.

For evaluation of IGSCC, those stresses that are present over long periods of time (such as those produced by pressure on restraint of thermal expansion) are probably more significant than those present over relatively short periods of time (such as those produced by temperature gradients or safety relief valve thrust). However, even though these stresses are of short duration, they may influence propagation of cracks because of their cyclic nature.

6.4.1 Code Limits on Operation Stresses

The magnitude of operation stresses is limited by the codes under which the piping systems are designed. For BWRs now in operation, the code used in design was ANSI B31.1.0, "Power Piping." The ASME Boiler Code, Section III, "Nuclear Power Plant Components," is the current code used for design of piping systems in nuclear power plants.

Design codes for piping use a rather complex set of stress limits, none of which includes allowances for metal deterioration such as IGSCC. A nominal "corrosion allowance" may be included and some warnings may be given (e.g., graphitization of steels at elevated temperatures), but, in general, the designer is expected to use materials that will not deteriorate in the anticipated environment. Also, design codes for piping do not consider

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(a) Shack, Ellingston and Pahis (6.4) include residual stress data on 4-in. pipe-to-elbow welds from piping that was removed from BWR service. The authors speculate that: "The relatively low (compared to pipe-to-pipe test welds) stresses seen in the straight-pipe-to-elbow weldments are most likely due to the different restraint imposed by the elbow geometry. The two elbow welds may also have been stress-relieved (annealed); however, metallurgical examination, while not conclusive, suggests that this is not the case."

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 O.L.
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the following documents are to be served on the persons indicated by an asterisk (*) by hand delivery on May 4, 1982, and to the remainder of the Service List by placing them in the mail, first class, postage prepaid, on May 4, 1982:

1. Testimony on behalf of Suffolk County on Suffolk County Contention 24 and SOC Contentions 19(c) and 19(d);
2. Testimony on behalf of Suffolk County on Suffolk County Contention 26;
3. Testimony on behalf of Suffolk County on Suffolk County Contention 28(a)(i) and SOC Contention 7.A(1);
4. Testimony on behalf of Suffolk County on Suffolk County Contention 28(a)(iii) and SOC Contention 7.A(3);

5. Testimony on behalf of Suffolk County on Suffolk County Contention 31;

6. State of Qualifications of Robert Neil Anderson and Vera Barlit.

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