RELATED CORRESPONDENCE

DOCKETED USNRC

'84 APR -2 A11:14

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	:	
Public Service Electric and Gas Company	:	Docket No. 50-354 O.L.
	:	
(Hope Creek Generating Station)	:	

THE PUBLIC ADVOCATE'S FIRST RESPONSES TO THE APPLICANT'S FIRST SET OF INTERROGATORIES

Contention 1

1. "Specify all applicable NRC regulations, general design criteria, guidelines or other regulatory requirements, or portions thereof, pertaining to the phenomenon of intergranular stress, corrosion, cracking("IGSCC") which intervenor asserts are applicable to the recirculation piping installed at Hope Creek."

Public Advocate response: The Public Advocate believes that the following NRC regulations and GDC are applicable to this phenomenon:

1. Criterion 1 -- quality standards and records ("A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records . . . shall be maintained . . . throughout the life of the unit.")

2. Criterion 2 -- design bases for protection against natural phenomenon

3. Criterion 4 -- environmental and missile design bases ("These . . . components shall be appropriately protected against dynamic effects, including . . . pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.")

4. Criterion 14 -- reactor coolant pressure boundary (". . . an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.")

5. Criterion 15 -- reactor coolant system design ("The design condition [must not be] exceeded during any condition of normal operation, including anticipated operational occurrences.")

6. Criterion 30 -- quality of reactor coolant pressure boundary

7. Criterion 31 -- fracture prevention of reactor coolant pressure boundary

8. Criterion 32 -- inspection of reactor coolant pressure boundary

9. Criterion 33 -- reactor coolant makeup

10. Criterion 34 -- residual heat removal

11. Criterion 35 -- emergency core cooling

12. Criterion 36 -- inspection of emergency core cooling system

13. Criterion 38 -- containment heat removal

14. Criterion 39 -- inspection of containment heat removal system

15. Criterion 44 -- cooling water

16. Criterion 45 -- inspection of cooling water system

17. Criterion 51 -- fracture prevention of containment pressure boundary

18. Criterion 54 -- piping systems penetrating containment

The Public Advocate believes that the above general design criteria are applicable to the Contention relating to the potential for pipe cracking. When and if others are identified, the Public Advocate will inform the applicant and parties.

Additional "regulatory requirements, or portions thereof" include the following: NUREG - 0313, ACRS Letter of August 9, 1983, SECY 83-267 (July 1, 1983), SECY 83-267A (August 11, 1983), SECY 83-350 (August 18, 1983), NUREG - 0785, NUREG - 0803, NEDO - 24342 (April, 1981) ("GE Evaluation in response to NRC request regarding BWR SCRAM system pipe breaks"). In addition, please see Danko and Stahlkopf, "Status of Research on Pipe Cracking in BWR's", 23 Nuclear Safety, 653 (December, 1982).*

2. "Specify each section of the Hope Creek Final Safety Analysis Report ("FSAR"), which intervenor asserts is relevant to the consideration of the phenomenon of IGSCC in recirculation piping and specify, to the extent applicable, any failure to meet the regulatory requirements set forth in response to interrogatory 1 above."

The Public Advocate believes, at a minimum, the following sections of the FSAR are applicable:

FSAR Section	FSAR Amendment No.
1. Table 1.3-8 - Significant Design Changes from PSAR to FSAR	0
2. 1.8.1.31 - Conformance to RG 1.31	0
3. 1.8.1.36 - Conformance to RG 1.36	0
4. 1.8.1.37 - Conformance to RG 1.37	0
5. 1.8.1.38 - Conformance to RG 1.38	0

* Copy attached

6.	1.8.1.39 - Conformance to RG 1.39	0
7.	1.81.144 - Conformance to RG 1.44	0
8.	1.12.1 - Unresolved Generic Safety Issues	0
9.	1.14.1.86 - Generic Licensing Issues - Stress Corrosion Cracking of Stainless Steel	1
10.	5.2.3.2.2 - BWR Chemistry of Reactor Coolant	0
11.	5.2.3.4.1 - Avoidance of Stress Corrosion Cracking	0
12.	5.2.3.5.1.2. - SRP Rule Review	0
13.	QR 250.2	2
14.	QR 281.10	2
15.	QR 281.12	2

6. "Specify what recirculation piping intervenor considers to be 'critical'."

The intervenor considers all recirculation piping which utilizes type 304 stainless steel as critical. Not only is the piping of critical importance, but the use of this type of steel creates susceptibility to IGSCC.

7. "Specify what critical recirculation piping has not been identified by applicants as susceptible to IGSCC."

The intervenor does not understand this question.

8. "Specify the basis upon which intervenor relies for the assertion that connections to the decay heat removal system are critical piping."

Since the decay heat removal piping is connected with the recirculation piping in various places, the joints where the connections are made are also critical to the extent that they are made of either carbon steel or type 304SS. The welds are especially critical. Where the welds join dissimilar metals -- e.g., carbon and SS304 -- there is added risk of IGSCC.

9. "Specify the regulatory requirements and/or any other basis for intervenor's assertion that all recirculation piping susceptible to IGSCC must be replaced."

Please see general design criterion 4 ("Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents . . . ") If the recirculation piping susceptible to IGSCC is not either replaced or other protective measures installed, then intervenor asserts that GDC-4 has not been met.

10. "Specify the conditions under which intervenor asserts replacement is feasible."

The condition where replacement is most feasible is prior to plant operation -- i.e., before the facility goes "critical".

11. "Specify all preventive measures intervenor asserts applicants should take prior to start-up."

Please see response to Interrogatory No. 9 In addition, the applicant should consider the use of an oxygen control system. The intervenor understands that the facility has been designed in a way to allow installation of such an oxygen control system at a later date. This should be investigated now and considered for installation prior to any decision on licensing for operation.

The intervenor believes that such a system has recently been installed in the Susquehanna Nuclear Station.

In addition, the intervenor believes that General Electric now recommends the injection of signficant quantities of hydrogen into the feed water as a chemical method of reducing IGSCC. This should also be investigated and considered.

12. "Specify the preventive measures intervenor asserts should be taken prior to start-up for each designated 'critical' component of recirculation piping but which have not yet been taken."

Please see response to Interrogatories

13. "Specify the deficiencies intervenor alleges exist in the applicant's system for identification of cracks in recirculation piping after start-up."

As the Advisory Committee on Reactor Safeguards has noted (see Public Advocate Exhibit 1), it is a"delusion" to expect that the manual use of ultrasonic testing ("UT") is a reliable technique. The ACRS has noted the lack of "repeatability" of the technique. It is highly dependent upon the individual ability, initiative, and preservance of the individual testor. Moreover, given the cramped quarters in which this would have to be done, combined with the "hot" environment, UT methods will be difficult to employ. It is, in short, more of an art than a science. Thus, all practical preventive measures should be incorporated prior to plant start-up. The applicant should err on the side of preventive action.

14. "Specify those inspection techniques, other than manual ultrasonic testing, which intervenor asserts

applicants should use to identify recirculation piping susceptible to IGSCC after start-up."

There appears to be a misunderstanding in this sentence. The intervenor asserts that all SS304 piping in the system is susceptible to IGSCC. Thus, no special testing is needed to determine which piping is susceptible. Either it uses SS304 or it does not.

Nevertheless, since UT is not made for this purpose, a method should be devised and employed to help focus inspection and testing of SS304 piping. The intervenor believes that the General Electric "SRI" method is a helpful and reasonably dependable method for targeting piping and welds where such cracking is most likely to occur. Please see response to Interrogatory No. 11

<u>Contention 2</u>

1. "Define 'management implications' as used in this Contention."

The Public Advocate defines "management implications" in the manner used by the Board in its Order of December 21,

1983 (see p. 9-11). Thus, the Board describes intervenor's concern with the "serious shortcomings in the administration, procurement, maintenance and quality assurance programs at the Salem [Nuclear] Station, all of which are under the direct responsibility of PSE&G." Id., at 9. Absent a clear demonstration by PSE&G that the company has fully rectified the management aspects of company operation -- which led to, contributed to, and in substantial part caused the Salem circuit breaker outages of February 22 and 25, 1983 -- PSE&G cannot be found technically qualified to operate a third and distinct nuclear reactor. See 10 C.F.R. part 50, sections 50.56 and 50.57(a)(4).

(The Public Advcoate does not wish to litigate the Salem outage question. The question presented is whether the company has "learned the lessons of Salem" -- to borrow a phrase -- and whether, having learned those lessons, the company has implemented a program to apply those lessons to the operation of the Hope Creek facility.)

By way of additional explanation of "management implication" the Public Advocate submits the testimony of Dr. Stephen H. Hanauer previously submitted into evidence in a rate proceeding before the New Jersey Board of Public

Utilities, in the matter of the Motion of PSE&G to reduce the level of the LEAC, Docket No. 831-25 (December 1, 1983). Dr. Hanauer found, for example, that

> beyond the specific PSE&G programs that were found to be deficient, the conditions disclosed by the investigations raised a general question regarding the overall performance and capability of PSE&G management at both plant and corporate level. The NRC staff concluded that two principal causes of the [outages] were (1) a perceived lack of resolve on the part of PSE&G managers and supervisors in enforcing adherence to procedures and (2) insufficient management involvement in establishing the safety perspective governing Salem operations. (Hanauer testimony, p. 6, lines 14-23) (emphasis added)

2. "Specify each and every aspect in which intervenor claims that PSE&G management in the administrative, procurement, maintenance and quality assurance programs for the Hope Creek Generating Station, as of this date, fails to meet all applicable regulatory requirements and license conditions imposed by the NRC."

The Public Advocate contends that PSE&G has not yet presented sufficient evidence that it has complied with 10 C.F.R. §50.57(a)(4), a burden which is that of the applicant to meet prior to any decision to grant an

operating license. The Public Advocate additionally contends that because of the management failures uncovered by the Salem investigations, special scrutiny in this area is warranted. (See, e.g., Hanauer testimony "the actions and omissions of PSE&G management contributed substantially to the [Salem] outage cause or duration." see, p. 45, l. 39-40)

Dr. Hanauer concluded by observing that "deficiencies were revealed in the areas of operations, training, surveillance, maintenance, procurement, control of vendor activities and information, and quality assurance." (see p. 47, l. 49-52)

He further concluded that "beyond the specific PSE&G programs that were found to be deficient, the conditions disclosed by the investigations raised a general question regarding the overall performance and capability of PSE&G management at both plant and corporate levels. . . . [including] insufficient management involvement in establishing the safety perspective governing Salem operations." (Id., at p. 48, l. 1-11)

3. "As to each alleged deficiency, specify the applicable NRC requirement and/or license condition and describe in detail:

a. the applicable NRC regulatory requirement or license condition;

Response: 10 C.F.R. part 50, §50.56 and 50.57(a)(4).

b. "the precise management functions alleged to be deficient;"

Response: The Public Advocate believes that this question is answered by the response to questions 1 and 2, above, particularly the testimony of Dr. Hanauer.

c. "the name and/or job titles of the particular PSE&G management officials with responsibility for preventing or eliminating the deficiencies alleged;"

Response: The Public Advocate believes that insuring the safe operation of a nuclear facility requires the active involvement and explicit commitment of the entire corporation -- from the Board of Directors to the shift supervisor and those they employ and control. As to the specific named "PSE&G management officials", the Public Advocate has not yet completed analysis of Chapter 13 of the Hope Creek FSAR and PSE&G's response to the Public Advocate interrogatory no. 2, p. 7, in which PSE&G set forth the names of 132 personnel involved in the reactor trip-breaker incident response. The Advocate will inform PSE&G of its response to this question as soon as possible.

> d. "the acts or omissions performed by such individuals, identified by name or title, upon which intervenor relies in asserting that management deficiencies exist, including the dates of occurrence;

Response: please see the response to question 1 and 2 above, generally, with a fuller response to follow.

> e. "the actions which should have been taken by such management officials identified above in order to prevent or eliminate the alleged management deficiencies;"

Response: the Public Advocate believes that a program of training and management sufficient to guarantee as much as humanly possible the independence, vigor, and commitment of officials related to Hope Creek operations is essential as a predicate to any start-up decision. The history of nuclear power operation is replete wit! instances of corporate management blurring the independent, safety responsibility of nuclear power operators. Thus, e.g., serious incidents and accident "precursors" have occurred at such facilities as Three Mile Island and Browns Ferry due to in-plant maintenance occurring during power plant operation. In the case of Salem, PSE&G was not vigilant in the maintenance of a safety-related circuit breaker device, and then chose to attempt a repair of the device rather than simply to discard

it and replace it with a new one. Then, the company attempted to restart the facility without doing an adequate analysis of the causes of the prior "trip." This series of events suggests that corporate management has not fully instilled a "safety first" attitude among all those with responsibility for the operation of the Salem Reactors. It is, therefore, incumbent upon PSE&G to demonstrate that Hope Creek will be operated differently.

4. "If intervenor contends that PSE&G is not technically qualified to engage in the activities to be authorized by an operating license for the Hope Creek Generating Station within the meaning of 10 C.F.R. §50.57(a)(4), specify and discuss in detail:

> a. the particular aspect as to which PSE&G lacks technical qualifications;
>
> b. the extent to which intervenors claim is based upon any portion of the record of the application and, if so, identifying the particular portions thereof;
>
> c. all actions which must be taken by PSE&G in order to eliminate any alleged deficiencies in its technical qualifications asserted above."

Response: The Public Advocate believes that PSE&G has not demonstrated technical qualification in respect to its commitment to the safe maanagement and operation of

the Hope Creek facility. At this time the Public Advocate does not assert that the individual operators and managers lack expertise, although, upon further review, this allegation may be raised. The Advocate intends to supplement this response -- particularly with respect to 4c -- in the near future.

5. "To the extent that intervenor relies upon the reactor trip circuit breaker failure at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983, including all investigative and enforcement actions undertaken by the NRC with respect to those events, specify and describe in detail:

> a. each particular finding or statement by the NRC upon which intervenor relies to establish a "management implication" for Hope Creek;

<u>Response</u>: a management implication for Hope Creek is found in every finding and investigation regarding the specific deficiencies in maintenance, procurement, information flow and vendor control. All of these are the responsibility of management. See also Hanauer testimony, generally.

> b. the regulatory requirement or standard to which such statement or findings specified above applies;

Response: for example, PSE&G management in the area of the safety classification of equipment("equipment evaluation") was deficient. The identification of cause of failure of the February 22 incident was also deficient. Verification testing following reactor trip circuit breaker maintenance or initial installation may also have been deficient. (The company used the wrong kind of lubricant; also, certain "tabs" were broken off the replacement circuit breakers at the time of installation -- but this failure was not noted because the replacement breakers were not inspected on arrival at the plant.) Moreover, the ATWS incident revealed many "human factors discrepancies" which included "lack of internal consistency, logical ordering of steps and convention used for emphasis." See, e.g., "NRC safety evaluation related to plant restart PSE&G, Salem Nuclear Generating Station," NUREG - 0995, at p. 14. Further problems were demonstrated in operator training, training on revised procedures (5 of the 7 trainees could not answer the test on revised procedures successfully, Id., at 16). In conclusion, the NRC approved restart at Salem based upon certain commitments with respect to Salem operation. The intervenor believes that clear assurances are needed for Hope Creek given the company's prior demonstrated difficulties and the distinctiveness of the Hope Creek reactor from the

Salem facility (i.e., Salem is a pressurized water reactor while Hope Creek is a boiling water reactor).

Contention 3

1. "Specify all appliable NRC regulations, general design criteria, guidelines or other regulatory requirements, or portions thereof, pertaining to environmental qualification of safety-related electrical and mechanical equipment, components and subcomponents at the Hope Creek Generating Station."

<u>Response</u>: The Public Advocate is aware that the law in this area (environmental qualification) is in a state of flux. Following the decision of the Court of Appeals in <u>Union of Concerned Scientists v. Nuclear Regulatory Commission</u>, 711 F.2d 370 (C.A.D.C., 1983). Thus, 10 C.F.R. part 50, §50.49 which suspended the June 30, 1982 deadline for "completion of environmental qualification of safety-related electric equipment" may no longer be applicable law. See also, 49 <u>F.R.</u> 8422 (March 7, 1984) (wherein the NRC discussed its reaction to <u>Union of Concerned Scientists v. Nuclear Regulatory Commission</u>, <u>supra</u>.) The Commission's statement of policy highlights the importance of an adequate "E.Q."

program being in place prior to the start-up of operation. See.e.g., 49 F.R., supra, at 8425:

> the environmental qualification of electrical equipment throughout a nuclear power plant to standards higher than those existing at the time the plant was licensed has proved to be a complex and difficult task. . . in many cases equipment can be replaced only when the plant is shut down. During such down time, licensees have many tasks to accomplish in addition to equipment qualification efforts. Delays may also result from the unavailability of qualified equipment and difficulties in testing existing equipment.

Furthermore, it is noteworthy that the NRC intends to authorize continued operation of facilities where equipment qualification has not been shown under a program of staff "justification for continued operation (J.C.O.)". The Commission noted that there are "persuasive technical and policy reasons why licensees' assertions and analyses may be relied on pending independent NRC staff review. The Commission notes that licensees received their operating licenses after extensive staff reviews including, in many cases, adjudicatory hearings." Id. at 8426.

Accordingly, current Commission policy favors a selective use of J.C.O. relying upon the untested assertions of utility operators -- specifically because of, "in many

cases, adjudicatory hearings" related to the operation of these facilities. A full-scale hearing in the E.Q. area, therefore, is important not only on its own merits, but as an adjunct to the above policy statement. (For additional discussion, please see the basis listed on pp. 15-16 of Appendix I to the Public Advocate submission of contentions on November 7, 1983".

2. "Specify each section of the Hope Creek FSAR including applicant's Responses to Staff Questions which intervenor asserts is relevant to the consideration of environmental qualification, safety-related electrical and mechanical equipment components and subcomponents of the Hope Creek Generating Station."

Response: On this point the Public Advocate relies on PSE&G's responses to Public Advocate interrogatories (dated February 14, 1984) at pp. 12-14, which lists 30 separate sections or subsections of the FSAR and FSAR amendments.

3. "Specify the deficiencies intervenor asserts to exist in the applicant's Environmental Qualification Program outlined in the FSAR and amphified in answers to staff questions."

Response: The intervenor asserts that the applicant has not demonstrated that safety-related electrical equipment will operate in the harsh conditions of an accident scenario and therefore there is a lack of adequate assurances at this point that the facility can be operated with an adequate margin of safety. The intervenor further asserts that it will be necessary to review the proprietary reports identified by PSE&G at pp. 17-18 of the applicant's response to the Public Advcoate's first set of interrogatories (February 14, 1984) in order to provide a definitive answer to this question. At any rate, the intervenor believes that the applicant and the Commission should be greatly concerned about the implications of Board Notification, 84-032 (February 13, 1984), "Additional Information on Environmental Qualification" (transmitting Scandia Laboratories Annual Report for fy 83 for the Environmental Qualification Inspection Program conducted by Scandia as part of the NRC's "vendor inspection program"). In this report, Scandia reveals some astonishing areas of deficiency in the vendor qualification program. Thus, at p. 5, item no. 2, "qualification strategies,

it is noted that certain unnamed vendors have engaged in the practice of changing testing parameters in order to reach a positive result. And, as far as the majority of items tested, these simply failed. [redacted]. In short, the intervenor[redacted] is concerned[redacted] that the applicant may be relying on "qualification strategies" by vendors which fail to reveal the true unreliability of critical equipment.

Equally fundamental, vendors appear to rely upon a strategy of testing which uses high doses of radiation to simulate environmental conditions. These tests overstate reliability, inasmuch as low doses tend to degrade equipment more than the high does. As a result, PSE&G may have been mislead in the qualification of electrical cables, switches, and other such equipment that is important to safety, especially in an accident situation. (See also, Board Notification 84-007 (January 12, 1984), and I.E. Bulletin 840-2 and 84-20.)

In summary, the utilities E.Q. program suffers at the outset from a lack of reliability due to the lack of reliability on the part of vendors' E.Q. testing.

The Public Advocate continues to review this material and will supplement this answer.

Status of Research on Pipe Cracking in BWRs

By J. C. Danko* and K. E. Stahlkopf†

Abstract: *Intergranular stress corrosion cracking of welded type 304 stainless steel piping in boiling-water reactors has reduced plant availability and increased operating costs. In response to this problem, a major 4-yr research program, jointly sponsored by a Boiling Water Reactor Owners Group and the Electric Power Research Institute, has been established. A model for the phenomenon has been developed, and engineering solutions have been identified, developed, qualified, and implemented. These include solution heat treatment of shop welds, application of corrosion-resistant clad, alternate pipe materials of 304 and 316 nuclear-grade stainless steels, induction heating stress improvement, and heat-sink welding. Solutions under development include last-pass heat-sink welding, the control of water chemistry by startup deaeration and hydrogen additions, and improved methods for the detection and sizing of cracks. A unique facility for the application of the solutions to full-size pipe mock-ups and training of utility and service organization personnel has been established to ensure the transfer of technology.*

Since 1974 the number of worldwide incidents of intergranular stress corrosion cracking (IGSCC) in the weld-heat-affected zones (HAZ) of American Iron and Steel Institute type 304 stainless steel (304 SS) in boiling-water reactors (BWRs) has been increasing. To date, such incidents total 272. These are listed by piping system in Table 1. Some of the pipes are shown in the schematic of a BWR recirculation piping system in Fig. 1. In the recirculation system, the pipe diameters range from 10 to 71 cm (4 to 28 in.) with ~150 to 200 welds, depending on the particular BWR model (models 1 to 6).

Of the worldwide IGSCC incidents reported, there has never been an occurrence of pipe severance. Pipe cracking has been detected by the presence of small leaks or by ultrasonic and dye-penetrant inspections. Of the 272 reported incidents, ~35% have been detected by leaks. On the basis of the field data and pipe integrity analyses, a concept of "leak-before-break" was evolved.[1] This view is accepted by both industry and a special Nuclear Regulatory Commission (NRC) Pipe Cracking Study Group (PCSG).[2] The concept is based on the fact that leaks should develop in the cracked HAZ before the ductile austenitic stainless steel loses its structural integrity. With acceptance of the leak-before-break model, IGSCC incidents were not considered to be a safety issue or to represent a hazard to the safety of the public.[2]

However, the IGSCC problem did reduce plant availability and increase costs. In response to this problem, the Electric Power Research Institute (EPRI) initiated research and development (R&D) projects in early 1975 to resolve the IGSCC phenomenon. Early results provided a model[3] for IGSCC of welded 304 SS and identified several engineering solutions[4] to the problem. However, the number of incidents continued to increase, and the phenomenon spread to lines of 61 cm (24 in.) in diameter.[5,6]

Because of the generic nature of the problem and the continued impact on the utility industry, the Boiling Water Reactor Owners Group (BWROG) was organized to provide additional funds to the EPRI budget to accelerate a solution.

This article (1) presents the chronological developments of IGSCC in BWRs, (2) covers the results of the early EPRI R&D activities, (3) reviews the formation of the BWROG, and (4) describes the present status of the program.

---

*Joseph C. Danko joined the Electric Power Research Institute (EPRI) in 1978 and is presently the program coordinator for the Boiling Water Reactor Owners Group research program on intergranular stress corrosion cracking of stainless steel piping. He received the B.S. degree in metallurgical engineering in 1951 from the Carnegie-Mellon Institute of Technology and the Ph.D. degree in metallurgical engineering from Lehigh University in 1955. He has held management positions in nuclear materials research and development at the Westinghouse Electric Corporation and the General Electric Company.

†Karl E. Stahlkopf is presently director of the Nuclear Systems and Materials Department of EPRI. He joined EPRI in 1973 and has served as both manager of the Pressure Vessel Technology Program and acting director in the Nuclear Systems and Materials Department. He received the Ph.D. degree in nuclear engineering from the University of California, Berkeley. He served in a number of positions related to nuclear submarine propulsion under Admiral H. G. Rickover, including 1 yr as a staff officer with the Chief of Naval Operations at the Pentagon.

Table 1 Worldwide Incidents of IGSCC (as of 1-12-82)

Pipe system	Number of incidents
Recirculation bypass	50
Core spray	82
Control rod return line	3
Reactor water cleanup	67
Large recirculation [≥25 cm (10 in.)]	18
Small lines [≤7.5 cm (3 in.)]	16
Others (such as isolation condenser)	36
Total	272

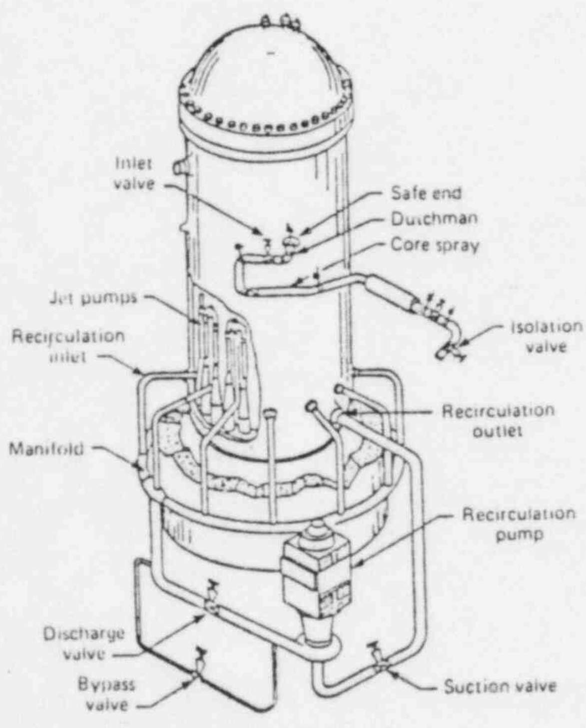

Fig. 1 BWR pipe system schematic.

HISTORICAL DEVELOPMENTS

Early Investigations of IGSCC

The phenomenon of IGSCC dates back to the operation of the first commercial BWR, Dresden Unit 1 (Ref. 6). Cracking incidents occurred early in the operational history of the plant. These included furnace-sensitized components that resulted from the postweld stress relief of the pressure vessel, as-welded pipe, and cold-worked pipe of 304 SS. In subsequent plants coming on line, IGSCC incidents continued. Although these presented problems to the utilities in the nature of repair and loss of plant availability, the phenomenon was not considered generic at that time.

Shortly after the first incidents of IGSCC, results of R&D work supported by the reactor vendor, General Electric (GE), began appearing in the open literature. The papers provided important technical information on the elucidation of the IGSCC phenomenon. In terms of a mechanism, one of the conditions for imparting susceptibility to 304 SS is the depletion of chromium in the austenitic grain boundaries that results when chromium carbides ($CR_{23}C_6$) precipitate during certain heat treatments and welding. This phenomenon, termed sensitization, alters the electrochemical behavior or the passivation of the grain boundaries. Passivation, or protection of the grain boundaries, is lost when the chromium content is <12 wt.%. This compares with a bulk content of ~18 wt.% in 304 SS. The chromium depletion models and theory are covered in a comprehensive review by Cowan and Tedmon.[7]

A film rupture model for the IGSCC initiation and propagation was also developed.[8] This model provided the basis for the current work on model refinements and improvements. The role of applied stress and dissolved oxygen content in the high-purity water on the IGSCC of furnace-sensitized 304 SS was determined.[9] In general, applied stress above the yield stress was required to produce IGSCC. Increasing the oxygen content above the 200 to 300 ppb steady-state value of the reactor accelerated IGSCC. On the basis of these laboratory results, furnace-sensitized 304 SS is no longer used in BWRs.[1] Studies on the effect of cold working showed 304 SS to be more susceptible to IGSCC, and this led to specifications on the control of cold work. Some effort was also devoted to screening tests of alternate materials that were more resistant to IGSCC, and a number of commercial alloys were identified. These studies made a significant contribution toward explaining the mechanism of IGSCC in furnace-sensitized 304 SS; however, the studies did not completely describe the behavior of welded 304 SS.

EPRI Response to the IGSCC Problem

The IGSCC incidents in weld-sensitized 304 SS showed a substantial increase in late 1974

in both domestic and overseas BWRs. This prompted the shutdown by the NRC of 15 BWRs[10] for inspection. All reactors were inspected and restarted. The NRC subsequently organized a special PCSG[2] to study the problem, report their assessment, and recommend corrective action. At GE,[1] a task force was also organized to address the IGSCC problem. In late 1975, EPRI initiated the first of a series of projects with the objectives of providing a better fundamental understanding of the IGSCC phenomenon and developing and qualifying engineering solutions for mitigation. The outputs of all these activities started to appear in late 1975, and, in general, the views of the phenomenon of IGSCC at that time have been confirmed by later work.

Stress analyses performed on the piping systems showed that the original calculations were within both the allowable values of the American National Standards Institute Pressure Piping B 31.1 and the allowable values of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Sec. III. One very significant finding of these studies was that no allowance was made in the code for weld residual stresses. If these stresses are included in the pipe design analysis, then some installations of 304 SS pipes exceed the code allowable values. Weld residual stress measurements performed on several different pipe sizes of 304 SS showed a pipe-size dependency; the peak residual stresses on the inside surface decrease with increasing pipe diameter.[11]

The weld residual stress was the missing link in the understanding of IGSCC behavior of 304 SS pipe. There was general agreement among the various organizations[1,2,11] as to the causative factors in IGSCC: (1) weld sensitization, (2) stress above the yield stress of the base material, and (3) a high level of O_2 dissolved in the cooling water. A model for the IGSCC of welded 304 SS was developed based on the coincidence of these three factors in the 304 SS pipe weldment. Weld residual stress was included as a major contributor to the total stress in the weldment.

On the basis of this model,[3,11] engineering solutions (commonly referred to as countermeasures or pipe remedies) for mitigating IGSCC in welded 304 SS pipes were identified.[4,12–14] The remedies can be logically divided into three categories: stress, sensitization, and coolant environment. Work was initiated on a number of remedies; their status is presented later (see BWROG Program).

The NRC Pipe Crack Study Group and EPRI also examined in-service inspection (ISI) methods and requirements.[2] The NRC ordered augmented ultrasonic inspections to be conducted in accordance with the ASME Code Sec. XI, but the specified schedule for examination was reduced on the basis of the specific piping system. The nondestructive examination practices for ISI were reviewed, and a number of problems were identified.[2] They were specifically related to (1) the weld preparation configurations of the counterbore and weld crown that give rise to a multitude of geometrical reflections, (2) the working conditions of radiation, (3) temperature and humidity, and (4) the lack of trained personnel. Improved ISI methods and equipment for improving the sensitivity or detection and location of IGSCC were also recommended. Work in these areas was initiated under EPRI support.

Formation of the BWROG

Although satisfactory progress on the project activities discussed in the previous sections was being achieved, two important IGSCC events occurred that changed the time schedule and scope of work. These events were the IGSCC in the Alloy 600 safe ends of Duane Arnold[6] and in a 61-cm-dia. (24-in.) 304 SS pipe in KRB Block A in Germany.[5] Both cases were linked small cracks, with the field of cracking covering the entire circumference of the pipe. Because of the extent of cracking, the NRC reconvened the PCSG.[6] The group concluded that further studies on pipe integrity were needed; however, the concept of leak-before-break remained valid.

These events also triggered action by the BWR utility owners to add additional resources to help solve this problem. In late 1979 the BWROG was established, and EPRI was requested to manage the owners group research program. Since the original group was formed, the program was expanded in 1980 to include overseas BWR utility owners, and 10 have decided to participate in the program. Total membership at this time is 32.

BOILING WATER REACTOR OWNERS GROUP IGSCC RESEARCH PROGRAM

EPRI accepted the responsibility to integrate the resources of the BWROG in the ongoing R&D

effort into a single unified program to meet the utility needs. This program is the BWROG IGSCC research program.[15]

The objective of this research program is to develop the technology to extend the service lifetime of piping and to minimize the outage times required for inspection and repair. The application and implementation of the resulting technology will improve plant availability and reduce the economic impact of IGSCC in BWRs. The results of the program are applicable to operating plants, plants under construction, and future BWRs.

The present 4-yr program (1980 to 1983) places emphasis on (1) analysis, inspection, and testing of large piping of the recirculation system, (2) use of full-size mock-ups to demonstrate field applicability of various qualified remedies, and (3) improvements in nondestructive examination methods to detect, locate, and size IGSCC. To address these needs, the program is structured into three major tasks: problem resolution, remedy development, and remedy application. The objectives of each task and its current status are discussed in the following sections.

Problem Resolution

The objectives of this task are to (1) develop improved capability for identifying those pipe welds that are most likely to suffer IGSCC, (2) provide a capability of locating and sizing IGSCC, (3) provide reliable methods for predicting and monitoring crack growth, and (4) evaluate the consequence of cracking from a systems viewpoint.

Status of Task. General Electric has developed a stress rule to identify pipe welds that are potential IGSCC candidates.[16] The stress rule is

$$\frac{P_m + P_b}{S_y} + \frac{Q + F(resid)}{S_y + 0.002E} = SRI$$

where P_m = primary membrane stress
P_b = primary bending stress
S_y = code yield strength
Q = sustained secondary stress
F = sustained peak stress
E = elastic modulus
$resid$ = weld residual stress
SRI = stress rule index

At SRI values of 1.0 and less, the probability of IGSCC is very small indeed. The application of the SRI to field cases does show that no IGSCC has been found at values <1.2. However, the stress rule has not been successful in identifying which welds are vulnerable with SRI values >1.2.

Deficiencies in the stress model include (1) no consideration for the level of sensitization, (2) residual stress distribution, (3) effect of postweld grinding, (4) service stress cycles, and (5) the actual operating stresses. These factors are being evaluated in studies to improve the current model. From these studies, a promising engineering model for crack initiation based on a film rupture mechanism of IGSCC is being developed. This engineering model includes both cyclic and static loads and has been successfully applied to reconcile the results of tests on constant-extension-rate laboratory specimens and cyclic loaded pipe test specimens. Further work is in progress to include the other factors known to affect IGSCC.

Once the vulnerable welds are identified, ultrasonic ISI equipment and procedures are needed to detect, locate, and size the IGSCC in the HAZs of the austenitic pipes. Under this program, a partially focused dual-crystal search unit has been employed and appears to be superior to alternate units for ISI. Because the unit is partially focused, different configurations are required for varying pipe sizes and schedules.

A "call confirmer" ultrasonic (UT) unit consisting of a pulsor-receiver and search unit, a coupled signal processing unit, and a manual transducer positioning jig (scanner) is available in a field prototype version and is presently undergoing procedural development testing. The call confirmer views the UT return signal and compares its features with stored characterizations of IGSCC, and a decision is made that disinguishes the signal as a crack or as a geometric discontinuity.

To assist in improved crack sizing and monitoring, pipe samples with IGSCC have been produced for training and qualification of procedure. A process has been developed to introduce IGSCC in welded 304 stainless steel pipes that simulate field failures. Use of these samples or standards will reduce the error in detection which results from the use of conventional calibration notches in standard specimens. Standard IGSCC specimens of 25-cm (10-in.) Schedule 80 pipes will be distributed to the BWROG members for training.

Detection, location, and sizing of cracks provide part of the information needed to predict the service lifetime of the austenitic stainless steel piping system. The other source of information is derived from analytical and experimental studies of crack growth. In the early IGSCC incidents in small pipes, crack propagation was rapid. Accordingly, IGSCC detection led to repair or replacement of the piping. Recent data have demonstrated that crack propagation rates are strongly influenced both by residual stress distribution and by weld sensitization. The weld residual stresses are a strong function of pipe diameter[11] with the tensile residual stresses on the internal diameter of the pipe, decreasing with increasing pipe diameter. In Fig. 2 the measured through-wall residual stress distribution of large pipes is shown. Cracks initiated at the inside surface will propagate only until they reach the region of compressive residual stress, where they will arrest. Thus forced outages to repair large pipes may not be necessary. Crack-growth verification experiments on large pipes subjected to simulated service conditions are in progress to check on the life predictions.

All IGSCC incidents to date have resulted in leak-before-break behavior. A complete circumferential crack must exceed 50% of the wall thickness for the service loads to extend the crack by ductile tearing. It is predicted that, before the cracked area reaches this size, the azimuthal variations of welding residual stress and material susceptibility in the weld HAZ, combined with the applied bending loads, will lead to asymmetrical crack growth and to the formation of a through-wall crack at some location. This prediction is borne out by field experience. To date, all the cracks caused by IGSCC which have penetrated the pipe wall have resulted in a leak and have been detected well before there was any chance of catastrophic pipe failure under normal service loads.

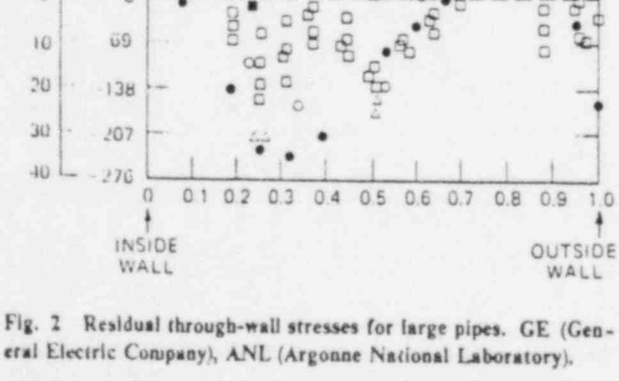

Fig. 2 Residual through-wall stresses for large pipes. GE (General Electric Company), ANL (Argonne National Laboratory).

Analytic and experimental studies in BWR austenitic piping have shown that stable crack growth caused by ductile tearing can only start after a critical plastic flow stress [about 4.8×10^8 Pa (70 ksi)] is reached in the pipe cross section. Delineating the transition between this stable tearing (leak) with increasing load and instability (break) is the critical aspect of the leak-before-break question addressed by current research on fracture of very ductile materials.

For structures made from tough ductile materials such as stainless steel, the occurrence of unstable fracture (which would lead to break-before-leak behavior) is now known to be determined by the dependence of fracture resistance (a material property) and crack driving force (which depends on load and structural geometry) on crack extension. Because laboratory measurements have determined how the fracture resistance of stainless steel varies with crack extension, predicting crack growth instability becomes a matter of predicting how the driving force varies as the stress corrosion crack extends through the wall and around the circumference of the pipe.

Under NRC sponsorship, a stability analysis has been performed for a circumferentially flawed stainless steel pipe subjected to severe displacement-controlled bending and axial load, a condition believed to be generally representative of seismic loading. The analysis showed that instability is unlikely if the pipe length-to-radius (L/R) ratio is <200. More recent work under the BWROG and work sponsored by the NRC has shown that L/R values for the piping lines studied to date have ratios well below 200. This is true, for example, for a circumferential crack 50% through the wall, containing a through-wall section of 120°. The example assumes an axial load that is 50% of the flow stress and a bending load sufficient to produce in the cracked section a fully plastic moment that is significantly larger than the ASME Code design allowable for normal operation and antici-

pated transients. Thus instability is very unlikely under a large load characterized by displacement-controlled bending and axial loads even in the presence of severe IGSCC.

The validity of the approach to structural instability taken by the NRC analysis was demonstrated in small-specimen tests. Further verification was recently obtained in a bend test on a flawed 10-cm (4-in.) stainless steel pipe. In this test, the specimen was spring loaded to simulate a length of 8.8 m (30 ft). The results confirmed the transition from stable to unstable crack growth predicted by the stability analysis. Additional stability tests are planned in 10-, 25-, and 40-cm (4-, 10-, and 16-in.) pipes with part-through and through-wall flaws subjected to combinations of internal pressure, bending, and axial tension to simulate piping loads under normal and abnormal operation. Other work seeks to (1) estimate the crack driving force and its variation with crack extension for a variety of geometries and loads and (2) develop and apply simple engineering design methods for determining equivalent L/R values for the complex piping systems found in BWRs.

The results of the BWROG and NRC work indicate that unstable crack extension would probably not occur in BWR stainless steel piping systems designed in accordance with the ASME Code even though severe IGSCC may be present. The more probable consequence is leak-before-break behavior. This follows from (1) field experience with real piping systems, (2) the fact that asymmetric weld-heat sensitization and bending loads suggest a preferential direction for crack growth, and (3) stability analysis and associated verification tests.

Remedy Development

The objective of this task is to develop and qualify remedies for the mitigation of IGSCC in the austenitic stainless steel piping.

Pipe remedies have been organized into three categories as shown in Table 2. Discussion of the remedies follows the categories in that table.

Table 2 Remedies for Mitigation of IGSCC in 304 SS Pipe

Remedy	Purpose
Sensitization Related	
Solution heat treatment of shop welds	To eliminate weld sensitization and residual stresses
Corrosion resistant cladding	To provide an IGSCC-resistant layer to protect the HAZ
Alternate pipe material	To eliminate weld sensitization
Stress Related	
Heat-sink welding (HSW)	To provide compressive residual stresses on the inside surface and partially through the wall
Induction heating stress improvement (IHSI)	To provide compressive residual stresses on the inside surface and partially through the wall
Last-pass heat-sink welding (LPHSW)	To provide compressive residual stresses on the inside surface and partially through the wall
Environment Related	
Startup deaeration	Reduce dissolved oxygen content before startup
Alternate water chemistry	Add hydrogen to feedwater to reduce steady-state oxygen content

Sensitization-Related Remedies. The development work and qualification of solution heat treatment (SHT) of shop welds,[17-19] the shop and field application of corrosion-resistant cladding (CRC),[17-19] and alternate material[14] pipe remedies have been completed. Qualification was achieved by testing welded pipes using each remedy and comparing the results with reference as-welded 304 SS (Ref. 20). A statistical test program was formulated to relate the time to failure of the laboratory specimens to field failures and thereby to demonstrate the capability of the pipe remedies to achieve the goal of a 40-yr plant lifetime.[21] For 10-cm-dia. (4-in.) pipes, the goal was to show a factor of 20 improvement of the pipe remedy over the reference condition. This was an extremely conservative estimate because the test conditions accelerated the IGSCC. These conditions were (1) high applied tensile stress [136% of the yield stress of the base material at 288°C (550°F)], (2) cyclic loading (0.7 cycles/h trapezoidal wave shape), and (3) high dissolved oxygen content (8 ppm) in the high-temperature [288°C (550°F)] high-purity water.[20] The sensitization-related pipe remedies have been accepted by the NRC in NUREG-0313 Rev. 1 (Ref. 22) and provide relief from the augmented ISI of nonconform-

ing pipe lines. The SHT, CRC, and alternate pipe material 316NG have been used in plants under construction and for the repair or replacement of pipes in operating plants worldwide.[23,24]

Under the sensitization-related remedies, studies are being performed on the phenomenon of low-temperature sensitization (LTS). The objective is to determine the extent of continued sensitization at the BWR steady-state operating temperature of 288°C (550°F) on the susceptibility of the welded 304 SS to IGSCC. These studies include stress corrosion cracking tests, fundamental studies on the phenomenon and modeling analysis to predict the long-term operating effects.

Results of these studies have provided the following understanding of LTS. In the weld HAZ of the 304 SS, precipitation of chromium carbides ($Cr_{23}C_6$) occurs at the austenite grain boundaries. Examination of welded 304 SS specimens exposed at temperatures ranging from 288 to 500°C (550 to 932°F) for various times revealed further sensitization in the HAZ. This was manifested by growth of the precipitated chromium carbides with further depletion of chromium at the grain boundaries. On the basis of thermodynamic considerations, the chromium in equilibrium with the chromium carbides may drop to low levels of 6 to 8 wt.% chromium, depending on the time and temperature. Scanning transmission electron microscopy measurements of the LTS specimens did show a lower chromium content in the grain boundaries relative to the as-welded condition.

The effect of the lower chromium content on IGSCC of the welded 304 SS specimens is being determined by constant extension rate tests. Tests on as-welded and as-welded plus LTS specimens at 288°C (550°F) in high-purity water with 8000 ppb dissolved oxygen show that the LTS condition is more susceptible to IGSCC. Both test time and total elongation are significantly less in the LTS treated specimens. However, tests performed in a water environment with 200 ppb dissolved oxygen, the steady-state operating level, showed essentially no difference between the two metallurgical conditions.

For the fundamental and modeling studies, Arrhenius plots of the stress corrosion cracking tests and sensitization measurements are being used. To date, the results show that, for certain heats of 304 SS, LTS will occur within the plant design lifetime. The 304 and 316 nuclear-grade stainless steels do not exhibit LTS because of the low carbon content (0.02 wt.% max.). Although normal grades of welded 304 SS are prone to LTS, this does not enhance the IGSCC susceptibility at the BWR steady-state operating conditions. Further studies are in progress to fully evaluate the LTS phenomenon.

Stress-Related Remedies. The heat-sink welding process involves the welding of the root pass and the first and second layer by conventional welding and then completing the welding while flowing, sprayed, or stagnant water is used inside the pipe. With this process, favorable compressive residual stresses are introduced on the inside surface of the pipe and partially through the wall. An additional benefit is the slight decrease in the level of sensitization relative to conventional welds. Test results have shown a factor of improvement greater than 15 over reference pipe welds,[17,19] providing no postweld grinding is employed. The grinding operation introduces high tensile residual stresses that overcome the compressive residual stresses of heat-sink welding. The process has been effective for repairs in operating plants and plants under construction.[24,25]

The IHSI (Inductive Heating Stress Improvement) process, conceived, developed, and implemented in Japan,[26-30] involves heating the outside surface of the pipe weldment to ~500 to 550°C (930 to 1020°F) while maintaining the inside surface temperature at 100°C (212°F) with flowing water. The temperature gradient across the wall thickness results in ~0.5% strain in the stainless steel and imparts a compressive residual stress on the inside surface and partially through the wall. This compressive stress reduces the propensity for IGSCC initiation and also may arrest the propagation of existing cracks.

Under the BWROG Program, process specifications have been developed using a section of a recirculation loop mock-up, and residual stress measurement on the joints has been performed.[31] In addition, tests on IHSI treated pipe and reference welded material are in progress. To date, the IHSI pipes show a factor of improvement greater than 10 over nontreated 304 SS pipes. At this time, a number of domestic utilities have expressed interest in applying the process both to plants under construction and to operating plants.

The LPHSW (Last Pass Heat Sink Welding) process involves the use of high heat input during

the final weld while water flows through the inside of the pipe. This process, like heat-sink welding and IHSI, results in compressive residual stresses on the pipe inside surface and partially through the wall. Preliminary development work has demonstrated compressive residual stresses in a 71-cm (28-in.) pipe. Work is in progress to establish the process parameters that will be verified by residual stress measurements. Tests to quantify the benefits of LPHSW are also planned.

Environment-Related Remedies. In most BWRs, oxygen levels during startup are ~8000 ppb. This compares with steady-state levels of 200 to 300 ppb at the operating temperature of 288°C (550°F). The variations of dissolved oxygen for a number of BWRs with normal and startup deaeration have been measured and are shown in Fig. 3. Plants with deaeration practices clearly show lower oxygen contents at low temperatures.[32,33] Because of the high oxygen content during startup, it has been postulated that this was a major contributor to the IGSCC of the 304 SS piping. However, the steady-state conditions also facilitate IGSCC and, at this time, the benefits of startup deaeration have not been established.

Under the BWROG research program, experiments are being performed that simulate the startup conditions of oxygen and hydrogen peroxide concentrations and stress as a function of temperature. Tests up to 190°C (375°F), which is the temperature where the dissolved oxygen content is near its steady-state value, showed no difference between normal and deaerated startup.[34–46] However, when the tests were extended to the full operating temperature of 288°C (550°F), the results were not the same. The aerated startup simulations resulted in IGSCC, whereas the deaerated startups did not produce IGSCC. It appears that something may happen at temperatures below 190°C (375°F) that affects the behavior of the 304 SS above 190°C. In assessing the effects of startup deaeration, the operating cycle of the BWR must be considered. The reactor spends most of its operating time in the steady-state condition. Although deaeration during startup is less deleterious than normal aerated startup, the contributions between steady-state and startup environments must be separated before the benefits of startup deaeration can be quantified. This is the objective of experiments currently in progress.

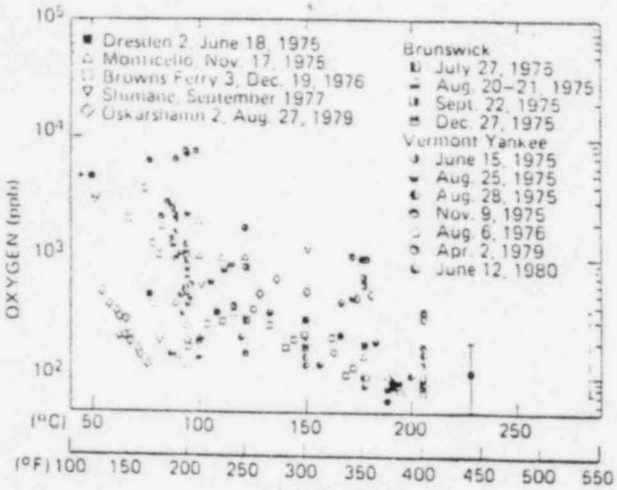

Fig. 3 Variation of oxygen with temperature during startup.

Alternate water chemistry involves the addition of hydrogen to the feedwater at operating temperature. Experiments performed in the Oskarshamn BWR in Sweden have demonstrated that small amounts of hydrogen can significantly reduce the dissolved oxygen content.[33] Electrochemical measurements of the corrosion potential of 304 SS electrodes showed a reduction. With sufficient hydrogen addition, the oxygen content was reduced to a level where the electrochemical potential would no longer support IGSCC.[37] Thus, by adjusting the oxygen content by hydrogen addition, IGSCC of 304 SS can be prevented. Present experiments are in progress to determine the effects of hydrogen water chemistry on IGSCC of 304 SS and other structural materials.[38] Under a joint Department of Energy and Commonwealth Edison Company project, in-reactor experiments are planned in Dresden 2 Power Station in 1982.

Remedy Application. The goals of this task are to (1) demonstrate the application of the various pipe cracking remedies on full-sized field mock-ups, (2) provide assurance that the remedies are effective, (3) train and qualify personnel in the application of the remedies, and (4) prepare generic specifications, procedures, and quality assurance plans that are applicable to field implementation.

Status of the Task. This task is assigned to the EPRI Nondestructive Evaluation (NDE) Center in Charlotte, N. C. The center includes the BWROG Pipe Remedy Application and Training

Facility. Construction of the center started in 1980, and it was dedicated in February 1981.

Several BWRs were surveyed to obtain information on the accessibility and fabrication of the piping systems. The surveys included one BWR-5 and three BWR-4 models with Mark 1 and Mark 2 containments. Access to potentially vulnerable pipe welds was more restrictive with the Mark 1 containment, but access in all plants presented problems. Piping system layout varied from one architect–engineering firm to another. Variations between the recirculation loops were also observed. This information was used in guiding the preparation of full-size pipe mock-ups.

Selection of mock-ups was based on field experience, results of a special study of repair/replacement of large pipe, and the survey studies. The mock-ups include (1) a discharge riser attached to both the ringheader sweepolet and the reactor pressure vessel inlet nozzle; (2) a recirculation loop discharge manifold (ringheader) and five discharge risers; (3) the lower horizontal section of the suction line with the downcomer elbow, a simulated suction valve, and the recirculation pump inlet; (4) a section of the suction downcomer with the residual heat removal tee in the vertical section; and (5) a simulated safe end on the reactor recirculation outlet nozzles (see Fig. 1). These mock-ups are in various phases of fabrication and are scheduled to be completed in 1982.

Various commercial pipe severing and machining equipment has been purchased and is being evaluated. This includes the determination of the merits and limitations of the equipment, adapting the equipment for the specific repair needs, and the preparation of operating procedures for the various planned repair activities.

Welding equipment, both manual and automatic, has also been procured and is undergoing evaluation. Procedures for the preparation and performance of various pipe welds are being developed. These will be used in the preparation of the mock-ups and demonstration of the pipe remedies.

Meetings have been held with the technical representatives of the utilities and a special industry advisory committee on the subject of training. Valuable information has been obtained on the level of training needed and the personnel that should attend the training sessions. Two levels of trainees have been identified: the maintenance supervisors and the operators. Training of the skilled craft was not recommended because the needs are very specific, and these can best be gained by the use of mock-ups at the plant site. The first training session for maintenance supervisors is scheduled for early 1982.

OTHER PIPE CRACKING MECHANISMS

Pipe Cracking from Fatigue

Fatigue cracking in BWR pipes results from vibrations or thermal cycling. Compared with the IGSCC incidents, fatigue cracking accounts for only a small percentage of the total pipe cracks in the BWR stainless steel pipe systems. A review of the Licensee Event Reports (LERs) revealed that 18 cracking incidents were fatigue related in pipes <10 cm (4 in.) in diameter.

In addition to these LERs, thermal-fatigue cracks have been reported in five BWRs in Sweden.[39] Cracking occurred in 25- and 40-cm (10- and 16-in.) 304 SS pipes that formed a branch connection between the feedwater and shutdown cooling systems. Large areas of cracking observed in the base material was caused by thermal fatigue resulting from temperature gradients when the feedwater of 180°C (356°F) mixes with the shutdown cooling water of 270°C (518°F) during normal operation. The 304 SS was not sensitized, and the cracking morphology was transgranular. Operational times of the plants were between 23 000 and 39 000 h.

Similar incidents were reported in a Finnish BWR in annealed 304 SS pipe sections where mixing of water of two different temperatures [280 and 130°C (536 and 266°F)] occurred.[40] The cracks exhibited a transgranular morphology. Corrective actions for the thermal and mechanical fatigue was to modify the pipe system design to minimize or eliminate the thermal gradients and vibrations.

Pipe Cracking in Cold-Worked (Nonwelded) 304 SS

A number of pipe cracking incidents have been attributed to the effects of cold work on the 304 SS.[41,42] In the cold-worked metallurgical condition, failures may be either transcrystalline or intergranular depending on the stress level, environment, and degree of sensitization.

The IGSCC incidents of cracking in 304 SS elbows in the shutdown cooling and cleanup system in Oskarshamn[22] presents an unusual and most interesting case. The elbows had a minimal carbon content of 0.05 wt.% and had been formed by cold bending with a resultant deformation of 15 to 20%. This deformation of the 304 SS resulted in the transformation of some of the metastable austenite to martensite. Measurements on the degree of sensitization revealed a complete absence of chromium carbide ($Cr_{23}C_6$) precipitation. Thus, with the exception of sensitization, the other conditions for IGSCC were present, namely, high residual stress and oxygenated water. A hypothesis for this IGSCC incident is proposed, based on the presence of some martensite that is known to lead to localized corrosion. When this corrosion penetrates the surface, initiation occurs. With a crack initiated, the probability of a crack intersecting a grain boundary would be high. The grain boundary, a source of impurity segregation, could set up acid crevice conditions in the BWR environment. The presence of the cold-work residual stresses would provide the driving force of crack propagation. Although the crack propagation may be slow, depending on the state of stress in the elbows, the operation time of 45 000 h would be sufficient for the crack to propagate through the wall and result in the leak that was observed in the 11-cm-dia. (4.5-in.) elbow. Although these results were unexpected, they do indicate that under certain conditions of environment, stress, and operating time, initiation and propagation of IGSCC may occur in unsensitized 304 SS. Very recent unpublished results on IGSCC in unsensitized 304 and 316 low-carbon stainless steels with fatigue precracks in a simulated BWR environment of 8 ppm dissolved oxygen content support the above hypothesis. These results suggest that other failures similar to those in Oskarshamn may occur in unsensitized cold-worked 304 SS in BWRs.

CONCLUSIONS

Results of the BWROG research program on intergranular stress corrosion cracking of stainless steel piping have led to the following conclusions:

1. The phenomenon of IGSCC in BWR stainless steel piping joints is now well understood.

2. IGSCC does not present a safety hazard if leak-before-break behavior is established. Experience shows that this has been the case in every one of the 272 field occurrences.

3. IGSCC of BWR stainless steel piping depends on three conditions being present simultaneously: high tensile stress, susceptible material condition, and environment.

4. The phenomenon is easily reproduced in the laboratory on prototype components.

5. A range of qualified mitigating procedures for plants under construction and new plants has been developed and qualified by extensive laboratory testing in the United States, Japan, and Europe. The mitigating procedures include replacement by alloys immune to IGSCC, SHT of joints after welding, corrosive-resistant cladding, heat-sink welding, and induction heating stress improvement.

REFERENCES

1. H. H. Klepfer et al., *Investigation of Cause of Cracking in Austenitic Stainless Steel Piping*, Report NEDO-21006-1, General Electric Company, July 1975.
2. Nuclear Regulatory Commission, *Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants*, NRC Report NUREG-75/067 (PB-246 645), NTIS, October 1975.
3. R. E. Hanneman, Models for the Intergranular Stress Corrosion Cracking of Welded Type 304 Stainless Steel Piping in BWR, paper 3 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.
4. H. H. Klepfer et al., Mitigation of Intergranular Stress Corrosion Cracking in Austenitic Stainless Steel Piping, Report IWG-RRPC, International Atomic Energy Agency, February 1978.
5. H. Flache and R. Ettemeyer, Piping Repair Case Study at KRB, Unit A, *Trans. Am. Nucl. Soc.*, 35: 487-488 (November 1980).
6. Nuclear Regulatory Commission, *Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants*, NRC Report NUREG-0531, NTIS, February 1979.
7. R. L. Cowan II and C. S. Tedmon, Jr., Intergranular Corrosion of Iron-Nickel-Chromium Alloys, in *Advances in Corrosion Science and Technology*, Vol. 3, M. G. Fontana and R. W. Staehle (Eds.), pp. 293-400, Plenum Press, New York, 1973.
8. D. A. Vermilyea, Film Rupture Model for Stress Corrosion Cracking, in *Proceedings of Stress Corrosion Cracking and H_2 Embrittlement of Iron Base Alloys*, R. W. Staehle, J. Hochman, R. D. McCright, and J. E. Slater (Eds.), National Association of Corrosion Engineers, 1977.
9. W. L. Clarke and G. M. Gordon, Investigation of Stress Corrosion Cracking Susceptibility of Fe-Ni-Cr Alloys

in Nuclear Reactor Water Environments, *Corrosion*, 29(1): 1-12 (January 1973).

10. Directorate of Regulatory Operation, *Failure in 4-in. Bypass Piping*, NRC Regulatory Bulletin 74-10, Office of Inspection and Enforcement, Sept. 19, 1974.

11. A. J. Giannuzzi, *Studies on AISI Type-304 Stainless Steel Piping Weldings for Use in BWR Applications*, Report EPRI-NP-944, Electric Power Research Institute, NTIS, December 1978.

12. R. E. Hanneman, Methods for Preventing and Ameliorating Cracks in BWR Piping, in *Proceedings of First U. S.-Japan Symposium on Light Water Reactor Materials*, R. Staehle (Ed.), National Association of Corrosion Engineers, in press.

13. S. Iwasaki, Methods for Preventing and Ameliorating Cracks in BWR Piping, in *Proceedings of First U. S.-Japan Symposium on Light Water Reactor Materials*, R. Staehle (Ed.), National Association of Corrosion Engineers, in press.

14. J. E. Alexander, Alternate Piping Alloy Qualification, paper 48 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 3, Electric Power Research Institute, May 1980.

15. R. E. Smith, J. C. Danko, and A. D. Rossin, BWR Intergranular Stress Corrosion Cracking Research Program Overview, *Proc. Am. Power Conf.*, 42: 757-762 (1980).

16. R. E. Hanneman and R. L. Cowan, *Mitigation of Stress Corrosion Cracking in Boiling Water Reactors*, presented at IAEA Specialist Meeting on Environmental Factors Causing Pipe Cracks and Degradation in Primary System Components, held in Vienna, Austria, Oct. 20-22, 1980.

17a. N. R. Hughes and A. J. Giannuzzi, *Evaluation of Near-Term BWR Piping Remedies, Final Report*, Report EPRI-NP-1222 (Vol. 2), Electric Power Research Institute, NTIS, November 1979.

17b. N. R. Hughes, *Evaluation of Near-Term BWR Piping Remedies, Final Report*, Report EPRI-NP-1222 (Vol. 3), Electric Power Research Institute, NTIS, May 1980.

18. N. Hughes, Solution Heat Treatment, Corrosion Resistant Cladding and Heat Sink Welding of 304 Stainless Steel Pipe Welds, paper 4 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

19. N. Hughes, Qualification of Solution Heat Treatment, Corrosion Resistant Cladding and Heat Sink Welding, paper 17 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

20. J. C. Danko et al., A Pipe Test Method for Evaluating the Stress Corrosion Cracking Behavior of Welded Type 304 Stainless Steel Pipes, in *Properties of Steel Weldments for Elevated Temperature Pressure Containment Applications*, p. 91, MPC-9, American Society of Mechanical Engineers United Engineer Center, New York, 1978.

21. R. Post, J. Lemaire, and W. Walker, Statistical Approach to Qualify Countermeasures, paper 15 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

22. C. Y. Cheng, R. M. Gamble, A. Taboada, and B. Turovlin, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*, NRC Report NUREG-0313, Rev. 1, Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, NTIS, July 1980.

23. J. P. Higgins, Implementation of Countermeasures, paper 51 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 4, Electric Power Research Institute, May 1980.

24. T. Kohima, TEPCO's Approach to Pipe Cracking of BWRs, paper 50 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 4, Electric Power Research Institute, May 1980.

25. R. Sasaki et al., Mitigation of Inside Surface Residual Stresses of Type 304 Stainless Steel Pipe Welds by Inside Water Cooling Method, paper 9 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

26. T. Shimizu, K. Enomoto, S. Araki, and T. Ikegami, Induction Heating Stress Improvement for Welded Pipe and Its Effectiveness, paper 10 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

27. N. Ohki, M. Araki, K. Wakasa, S. Nakayama, and A. Tanaka, Actual Proof Test of Induction Heating Stress Improvement for Large-Diameter Pipe, paper 11 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

28. G. Yagawa, Y. Hayase, T. Shimizu, Y. Sakaguchi, S. Nakayama, and T. Umemoto, Analytical Studies on Induction Heating Stress Improvement for Precracked Pipe, paper 12 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

29. S. Tanaka and T. Umemoto, Residual Stress Improvement by Means of Induction Heating, paper 13 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

30. T. Futami, T. Matsumoto, S. Inasaki, and T. Umemoto, Implementation of Induction Heating Stress Improvement to Plants, paper 52 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 4, Electric Power Research Institute, May 1980.

31. D. C. Bertossa, Induction Heating Stress Improvement Qualification, paper 16 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.

32. N. Ohki, Oxygen Control Operation During BWR Startup, paper 53 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 4, Electric Power Research Institute, May 1980.

33. P. Fejes, Deaeration Practices in Swedish BWRs, paper 54 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 4, Electric Power Research Institute, May 1980.

34. P. Andresen and F. P. Ford, Technical Considerations for Startup Deaeration Procedures to Minimize Stress Corrosion Cracking of 304 Stainless Steel Piping, paper 7 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 1, Electric Power Research Institute, May 1980.
35. P. Andresen, The Effects of Surface Preparation, Stress and Deaeration on the Stress Corrosion Cracking of Type 304 Stainless Steel in Simulated BWR Start-Up Cycles, paper 39 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 3, Electric Power Research Institute, May 1980.
36. M. Indig, Controlled Potential Simulation of Deaerated and Nondeaerated Startup, paper 40 in *Seminar on Countermeasures for Pipe Cracking in BWRs*, Report EPRI-WS-79-174, Vol. 3, Electric Power Research Institute, May 1980.
37. B. M. Gordon and M. E. Indig, Progress Toward Oxygen Suppression in the BWR, *Trans. Am. Nucl. Soc.*, 35: 491-492 (November 1980).
38. B. M. Gordon, W. L. Clarke, M. E. Indig, A. E. Pickett, and M. T. Wang, Effectiveness of Hydrogen Additions for Mitigation of Stress Corrosion Cracking in the BWR Environment, paper 20 in *Corrosion 81*, National Association of Corrosion Engineers, 1981.
39. D. Burkhart, *Thermal Fatigue Cracking in Swedish BWR Feedwater Systems*, presented at IAEA Specialists Meeting on Environmental Factors Causing Pipe Cracks and Degradation in Primary System Components, held in Vienna, Austria, Oct. 20–22, 1980.
40. H. Hanninen and J. Hakala, Pipe Failure Caused by Thermal Loading in BWR Water Conditions, *Int. J. Pressure Vessels Piping*, 9(6): 445-455 (November 1981).
41. L. Frank, W. S. Hazelton, R. A. Hermann, V. S. Noonan, and A. Taboada, *Pipe Cracking Experience in Light-Water Reactors*, NRC Report NUREG-0679, Nuclear Regulatory Commission, August 1980.
42. J. Bergstrom, *Recent Experience With Pipe Cracking in Swedish BWRs*, presented at IAEA Specialists Meeting on Environmental Factors Causing Pipe Cracks and Degradation in Primary System Components, held in Vienna, Austria, Oct. 20–22, 1980.

State of New Jersey

Department of Energy

Board of Public Utilities

In the Matter of the Motion of

Pubic Service Electric and Gas Company

To Reduce the Level of the

Levelized Energy Adjustment Clause

Docket No. 831-25

Testimony of Stephen H. Hanauer

for the

Office of Public Advocate

State of New Jersey

December 1, 1983

Technical Analysis Corporation
Arlington, Virginia 22207

SECTION 1. INTRODUCTION

Q. Please state your name and business address.

A. Stephen H. Hanauer, 4620 Dittmar Road, Arlington, Virginia 22207.

Q. By whom are you employed?

A. I am Vice-president of Technical Analysis Corporation.

Q. For whom are you appearing?

A. I am testifying at the request of the Public Advocate of the State of New Jersey.

Q. What are your qualifications to testify as an expert?

A. A personal resume is given in Exhibit SHH-1 attached to my testimony. I have testified in rate cases before the Public Service Commission of the State of Maryland and the Public Utilities Commission of the State of Delaware. I have also testified before the Public Service Commission of the State of Indiana in a rate base case and performed consulting services for the Public Utilities Commission of the State of California in another rate base case.

Q. What experience have you had related to reactor protection systems and failures of such systems?

A. As part of my experience in nuclear power plant safety and operations, I participated in the design, construction and operation of reactor protection systems at Oak Ridge National Laboratory. My work at the U.S. Atomic Energy Commission and the Nuclear Regulatory Commission included personal contributions and supervision of the work of others in this area. I was a principal author of NUREG 0460, "Anticipated Transients Without Scram for Light Water Reactors". As Assistant Director for Plant Systems, I supervised the Instrumentation and Control Systems Branch, which performs licensing reviews of reactor protection systems. At the time I left the Nuclear Regulatory Commission in December 1982, I was a member of the steering group for the rule making related to anticipated transients without scram events.

Q. Dr. Hanauer, what is the purpose of your testimony.

A. To present my technical review of the causes of the transients that occurred at Salem Unit 1 on February 22 and 25, 1983. The reactor protection system failed in both events to function automatically to shut down the neutron chain reaction as designed and as required. An extended outage of Salem Unit 1 resulted. I will also present the conclusions I have drawn from my evaluation.

order. Public Service Electric and Gas Company did not have the system they needed to investigate thoroughly and systematically all abnormal events.

<u>Purchasing</u> - Public Service Electric and Gas Company's purchasing program was not properly managed. They bought circuit breakers and spare parts using incorrect specifications, and without the required quality assurance checking.

<u>Managing the Work of Contractors</u> - Public Service Electric and Gas Company relied on Westinghouse Electric Corporation for design and manufacture of the Reactor Protection System, including the circuit breakers, for technical maintenance information and on at least one occasion for repair work, but did not make sure that the work performed and the information furnished were adequate. In fact, some of the Westinghouse work and information were not adequate. The lack of adequate management control by Public Service led to failure to detect and correct the problems.

<u>Quality Assurance</u> - There were specific failures in the Salem quality assurance program. In addition, the Public Service program showed a general lack of the necessary aggressiveness and inquisitive investigation of plant performance.

<u>Management</u> - Besides the specific management failures, the Salem events showed that Public Service Electric and Gas Company was deficient in its overall management of Salem operations. The U.S. Nuclear Regulatory Commission required an extensive corrective action program by Public Service and fined them $850,000 for the violations of regulation and license conditions associated with the failures.

Q. What are your overall conclusions regarding this outage?

A. I have concluded that:

1. <u>The outage extension at Salem Unit 1, starting February 25, 1983, and lasting at least until May 8, was caused by PSE&G mismanagement resulting in two failures of the Reactor Protection System</u>. (See pages 9-11)

2. <u>The failures of the Salem Unit 1 Reactor Protection System of February 22 and 25, 1983, revealed management deficiencies that induced the NRC to require the plant to remain shut down for an extended period. Restart was conditioned by the NRC on development by PSE&G of an extensive remedial program directed at the deficiencies, and accomplishment of all short-term tasks of this program</u>. (See Exhibit SHH-5)

The level dropped to the set point for initiation of an automatic reactor shutdown. The safety shutdown function is provided upon loss of water in the steam generator because this water is counted on to be available to remove the heat from the reactor core in certain kinds of emergencies. The low water level caused a shutdown signal to be sent to the circuit breakers that are supposed to interrupt the power to the control rod drive mechanisms, and so cause the rods to drop into the core shutting down the neutron chain reaction. Two such circuit breakers were in service but both failed to open. The rods were not released and inserted into the core by this signal.

Q. Was the reactor shut down?

A. Yes. The plant staff decided independently and correctly to shut down the reactor manually. After a momentary difficulty in which the handle of the manual shutdown switch was pulled out of the switch by the operator, the manual shutdown was successfully initiated 3.6 seconds after the automatic signal which had failed to initiate a shutdown.

Q. What then was the failure?

A. The automatic reactor shutdown system is supposed to initiate rapid shutdown of the reactor core upon the occurrence of any of a number of reactor and plant conditions for which shutdown is the prudent, safe, required action. The failure of the reactor protection system to shut down the reactor, in the presence of an actual signal showing that one of the plant variables was actually in the range calling for automatic shutdown, constituted a safety system failure. This was a breach of one of the lines of defense which provide for nuclear power plant safety at Salem and at all nuclear plants.

Provision of the automatic system, backed up by the manual operator action, is typical of the redundancy built into nuclear power plant protection systems. Either the manual or the automatic action can shut down the reactor core. If one fails the other will, with high reliability, take care of the problem. But maintaining the reliability of such a redundant system requires careful attention to maintaining reliability of each of the sub-systems which comprise the redundancy. There is the additional point that the automatic system is substantially faster than the human operators can be expected to function, and therefore provides an additional margin of safety compared to manual action.

The Institute of Nuclear Power Operations (INPO), funded by electric utility companies, issued a "Significant Operating Experience Report" that contains the following evaluation of the significance of the failures:

The water level in one of the steam generators was allowed to fall below the safety trip level by the operator exercising manual control over feedwater flow. The reactor protection system was still inoperative and the automatic reactor shutdown called for by the low water level in the steam generator again failed to take place. This time the discrepancy was immediately recognized by the plant operating staff, which suspected a false alarm. However, they quickly reviewed the values of the plant variables and initiated, 25 seconds after the unsuccessful automatic shutdown demand, a manual shutdown which was successfully accomplished.

Q. Why was there a prolonged plant outage?

A. Because of safety concerns. The failures of the reactor protection system to function on the occurrence of a shutdown signal on two occasions, and the plant staff's failure to identify correctly one of these failures before clearing the plant for restart, raised serious safety concerns.

PSE&G's concern was expressed by Mr. Richard Eckert, Senior Vice-president of Public Service Electric and Gas Company. On March 15, 1983 he stated before the NRC Commissioners:

> One thing that has not been emphasized in the presentations made by us to your staff or to you by the staff is the ultimate responsibility of the Company for the safe operation of the Salem station. We do not take such responsibility lightly.
>
> The controlling activity toward restart is satisfying the senior management of Public Service Electric and Gas Company, that the problem in the plant was an isolated occurrence and not indicative of overall management deficiency.
>
> Until this requirement is satisfied, the Public Service management will not allow the restart.
>
> (Please see Exhibit SHH-2, pages 14-15)

On March 24, 1983, Mr. Robert I. Smith, Chairman of the Board of Public Service Electric and Gas Company, stated to the Commissioners,

> In earlier testimony Mr. Eckert stated that we will not allow Salem to restart until we are certain that any problems surfaced by our review have been

On October 28, 1983, Public Service paid the civil penalty of $850,000.

Q. Dr. Hanauer, why do you believe this Federal action to be significant to the New Jersey Board of Public Utilities?

A. The NRC enforcement decisions and actions, give an independent appraisal of Public Service Electric and Gas Company's management of the Salem Station by the Federal agency charged with the regulation of nuclear power plants and knowledgeable about their operation and management. In my testimony, in Section 9 on management, I will refer to the detailed discussions and evaluations of PSE&G management by the NRC.

SECTION 3. REACTOR PROTECTION SYSTEM

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. In this section, I give a brief description of the system in which the failure occurred and discuss its importance to safety and plant operation.

Q. How is the neutron chain reaction controlled?

A. The neutron chain reaction in the nuclear reactor core is the source of energy in a nuclear power plant. The reaction is controlled by introducing or removing neutron absorbers into the core. This is done in two ways. The first way, used for slow changes, is to dissolve some boric acid in the water which is pumped through the core to cool it. The boron in the boric acid is a neutron absorber. By varying the amount of boric acid dissolved in the water, it is possible to control the neutron chain reaction. This method of control affects the core uniformly throughout its volume.

The second method of controlling the neutron chain reaction is to insert or remove long slender metallic control rods which contain a neutron absorbing material. They are moved up and down by an electrical drive mechanism so that more or less of the absorber is actually in the reactor core region. The pattern of control rods inserted into the various parts of the core can be used to control one or another part of the core, or the rods can be moved together to control the core more nearly uniformly.

Q. How is the safety function accomplished?

A. If rapid shutdown of the reactor core is required, the control rods are inserted from above so that all the rods go all the way into the core. In this position, they have the maximum shutdown effect. The rod insertion is initiated by

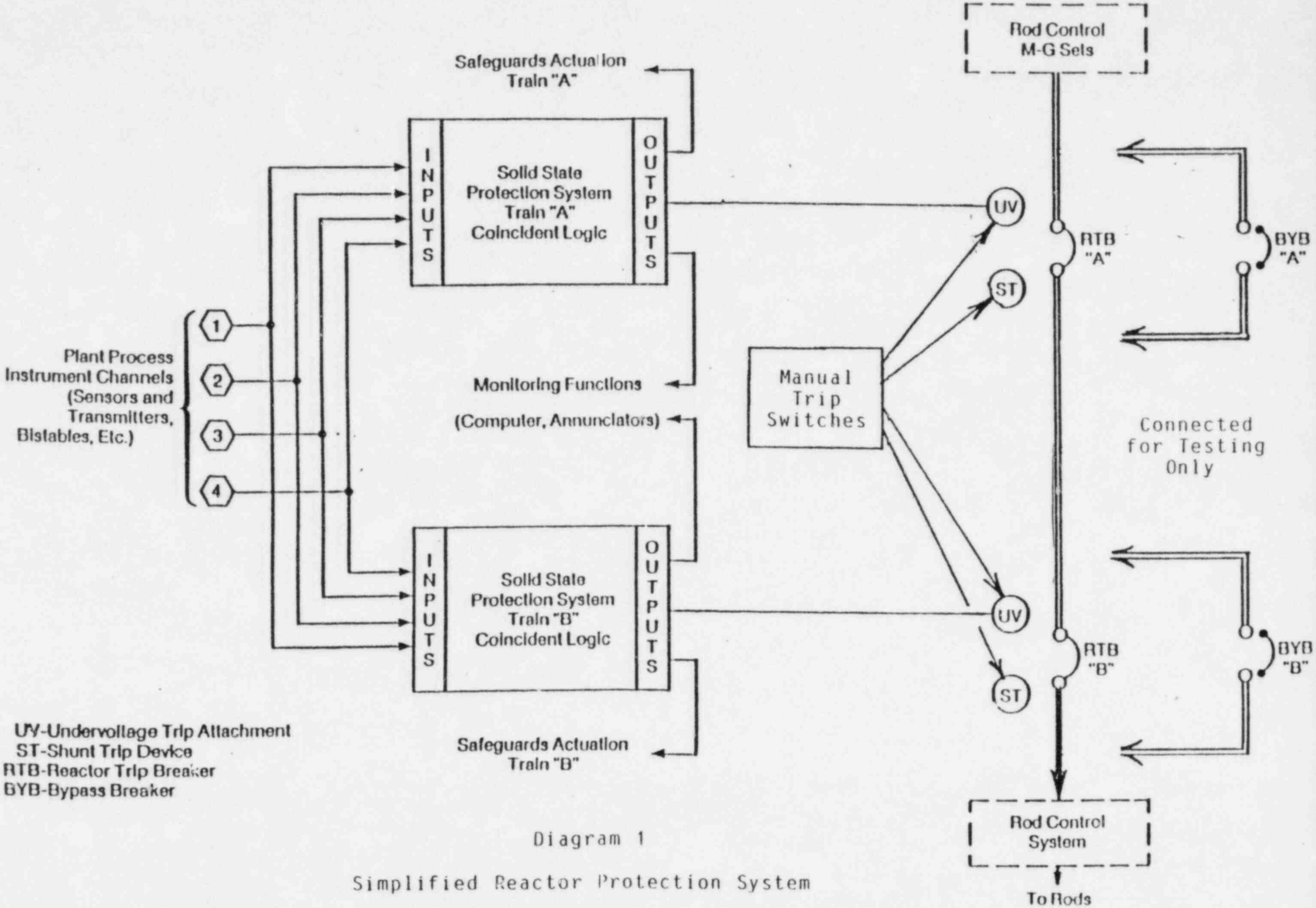

Diagram 1

Simplified Reactor Protection System

Q. Are these precepts embodied in regulations, codes or standards?

A. Yes. The basic standard governing design of reactor protection systems was issued by the Institute of Electrical and Electronic Engineers (IEEE) as IEEE-279. This standard has been adopted by the NRC in the Code of Federal Regulations Title 10, paragraph 50.55e. More recently the IEEE has issued other standards elaborating on the basic principles enunciated in IEEE-279 and discussed just above in my testimony. There is substantial literature on reactor protection system reliability, including the effect of common mode failures.

Q. Why the emphasis on common mode failures?

A. Because in well designed systems, common mode failures are much more probable than any other mode of failure in which the system function becomes unavailable. In the system we are discussing, provision of the two circuit breakers, either one of which can perform the system function, plus adequate testing, maintenance and repair, is expected to provide a high system reliability; that is, a low probability that the system function will not be available. Both circuit breakers have to fail for the system function to be unavailable. This can happen in two basic ways. The first is a common mode failure. The second is a failure of one circuit breaker, plus a concurrent but independent failure of the second circuit breaker. With reliable equipment and a good program of in-service testing, maintenance and repair, the second way can be made sufficiently improbable that common mode failures will be the principal way in which these systems fail. This is borne out by experience. If we count the Salem event as a single failure (which was not detected on February 22 but was detected on February 25), there have been three known reactor protection system failures in nuclear power plants in the Western countries. All three of these, including Salem, have been common mode failures.

Q. What is the importance of the reactor protection system?

A. The reactor protection system is one of the primary safety systems of the nuclear power plant. It is counted on to shut down the nuclear chain reaction when plant conditions exceed the allowable range and a shutdown is needed to place the plant in a safe condition. The reactor protection system, because of its central role in shutting down the power generation of the reactor core, must be highly reliable. The safety design basis of Salem, as for all nuclear power plants, includes the successful operation of the reactor protection system when it is needed and required. This requires that system failure be made so improbable that analysis of the plant with a failed reactor protection system is not necessary.

> misadjustment. These four categories are considered to fall under one broad category of maintenance related causes.
>
> (Exhibit SHH-9, page 59)

The ten findings of the Franklin Institute are given in Exhibit SHH-9 page 33. The failure causes are given on page 53 which discussed two possible failure modes. The first involves "latch-to-latch pin binding prevents unlatching of the UVT attachment thereby preventing the trip lever from moving when the device is de-energized." The second involves reduced force available to trip the circuit breaker as a result of manufacturing or adjustment errors.

During the visit I made to the Salem Station on November 16, 1983, plant personnel stated that one of the undervoltage trip attachments that failed on February 25, 1983, was observed by them shortly thereafter to "hang" in the energized postion even when it was de-energized.

At a public meeting of the NRC Commissioners on April 16, 1983, Dr. Zenons Zudans, a vice-president of Franklin Institute, summarized the results. The relevant pages of the transcript of this meeting are given in Exhibit SHH-11. Dr. Zudans stated:

> We concluded after Gary's examination that the most probable failure mechanisms are due to wear aggravated by lack of maintenance.
>
> (Exhibit SHH-11, page 12)

These two descriptions of the cause of failure are consistent. Mr. Gary Toman of the Franklin Institute told the Commissioners, at the same meeting:

> We had a meeting on Friday of last week where Public Service, Westinghouse, the NRC and Franklin Research met at Franklin Research Center. We discussed it. Westinghouse presented their findings. Franklin presented their findings. We determined if there was any significant variations and there were not.
>
> There was no debate concerning one set of findings versus the other. We did find out a little bit more about each other's research effort and why there seemed to be disagreements early on and they went away during the meeting.
>
> (Exhibit SHH-11, page 14)

Q. Why was the wrong lubricant used in January 1983 and why was no lubrication performed before that?

A. The reason for the failure to lubricate the breakers throughout their life at Salem and the reason for the use of an incorrect lubricant in January 1983 are really the same. PSE&G did not have the correct information regarding the lubrication needs of the breakers. Mr. Johnson has explained the facts which underlie this situation. (Johnson testimony pages 6-14) Originally, Westinghouse recommended that no lubrication be used on this type of circuit breaker. (Johnson testimony page 6) Later, in 1974, Westinghouse issued a technical bulletin recommending cleaning and lubricating these breakers as a result of experience with malfunctioning of a reactor trip breaker in another plant. One month later, also in 1974, Westinghouse issued another document which recommended a different lubricant. (Johnson testimony page 7)

Q. Why didn't PSE&G have the updated maintenance information from Westinghouse?

A. The answer to this question is not known. PSE&G states that Westinghouse has not been able to show that they sent the documents to PSE&G. (Johnson testimony page 7) My evaluation of the management by PSE&G of Westinghouse supplied information is discussed later in my testimony in Section 7, where I conclude that PSE&G mismanagement of the work of contractors contributed significantly to the outage.

Q. Why was the wrong lubricant used in January 1983?

A. The wrong lubricant was used because PSE&G mismanagement resulted in nobody on the job knowing the right one to use.

The correct lubricant is called 53701GW. Originally, Westinghouse specified no lubrication. Later, in 1974, Westinghouse recommended two lubricants that are no longer available. The lubricant actually used was Calfonex 78A, which is no good for the purpose.

Information regarding the service work performed in January 1983 was furnished by Mr. Johnson. (Johnson testimony pages 8-11) More detailed information was furnished by PSE&G to the NRC in a letter dated April 22, 1983. Please see Exhibit SHH-12, which also enclosed the Westinghouse Electric Corporation Service Report for the work performed by Mr. Esposito, a Westinghouse employee, beginning January 13, 1983.

The PSE&G letter states that the Westinghouse service man used "Calfonex 78A", which he happened to have a can of, rather than the correct lubricant. A further complication arose from the fact that the "correct" lubricant recommended in the second 1974 Westinghouse document is no longer avail-

Subsequent to the manufacture of the original Salem reactor trip breakers, as a result of the malfunctioning of a reactor trip breaker at another plant, Westinghouse modified the design of the reactor trip breaker undervoltage trip attachment. These modifications are specified in a Westinghouse document, NCD-Elec-18 of Dec. 17, 1971. The Westinghouse document specified that the style number will be the same for modified and unmodified undervoltage trip attachments. Undervoltage trip attachments are available with or without the 1971 modifications under the same type number. Only by specifying that the parts should conform to NDC-Elec-18, can one be assured that modified parts will be supplied. (Exhibit SHH-20, page 26)

PSE&G documents indicate that replacement undervoltage trip attachments were installed and tested in accordance with the Westinghouse document for all eight reactor trip breakers and trip by-pass breakers for Units 1 and 2 at Salem. The compliance document is dated August 8, 1972. (Exhibit SHH-21, pages 20-22)

PSE&G has given inconsistent statements on whether they were aware of the existence of NCD-Elec-18. PSE&G apparently told the NRC investigators that they were unaware of the existence of the modification or of the Westinghouse document that specified the modification or even that it had previously been implemented in August 1972. PSE&G stated that both NCD-Elec-18 and the compliance document were given to PSE&G by Westinghouse representative after the February 1983 events. (Exhibit SHH-21, page 21)

More recently, PSE&G has stated to the NRC in a letter dated October 28, 1983, that Westinghouse sent NCD-Elec-18 to PSE&G on January 26, 1972. (Exhibit SHH-22, page 6) Whether or not they received NCD-Elec-18 from Westinghouse in 1972, PSE&G purchased new undervoltage trip attachments in 1982 and 1983 without reference to the modifications specified in Westinghouse document NCD-Elec-18. (Exhibit SHH-21, page 23)

A Franklin Institute representative told the Commissioners on April 26 that one undervoltage trip attachment that he had studied did not appear to be manufactured in accordance with the changes of Westinghouse document NCD-Elec-18. (Exhibit SHH-11, page 16)

Again I refer you to Section 7 in my testimony for a more detailed discussion of PSE&G mismanagement of contractor activities and information.

Q. Aside from questions of lubrication, how good was the preventive maintenance program for the reactor trip breakers at Salem?

Q. Please discuss repair and corrective maintenance on the Salem reactor trip breakers.

A. After the events of February 1983, detailed inspection of the circuit breakers showed that improper adjustments and deformed and missing parts, ascribed to inartful repair work, were a factor in the failures. These anomalies are described in the reports by Westinghouse, (Exhibit SHH-9, pages 57-9) and by Franklin Institute (Exhibit SHH-9, pages 33 and 54; see also Exhibit SHH-11, pages 15 and 18)

Franklin Institute concluded that field adjustment or repair should not be attempted. Dr. Zudans told the Commissioners:

> The device does not allow any hardware modification on site. The only thing you are allowed to do is to lubricate in accordance with prescriptions. When it begins to fail, that is the end of life. You replace it. Whether the life is one year, two years or ten years, we do not know at this time because further tests are necessary for that.
>
> (Exhibit SHH-11, page 17)

Q. Were the reactor trip breakers kept clean and free of contaminants?

A. The breakers and the cabinets in which they are mounted were allowed to accumulate a great amount of dust. This is evident in the Westinghouse service man's report (Exhibit SHH-12, page 7) I believe that this shows lack of adequate attention and an unacceptable slovenliness.

Q. What do the maintenance history records show about repairs to the individual pieces of equipment?

A. PSE&G maintenance history records are very difficult to use for diagnosis or analysis. This means that a very important use of maintenance history was not possible at Salem; that is, evaluation of equipment maintenance history for items showing unusual failure rates or degradation trends, thus needing attention before possible failure.

Traceability of equipment, needed for adequate maintenance history records, was not maintained as breakers were moved to different positions and undervoltage trip attachments were moved to different breakers. This practice probably resulted in placing untested or inadequately tested equipment in service. The lack of traceability was a violation of NRC regulation (Exhibit SHH-6, page 9; Exhibit SHH-7, page 16)

Since these equipments presumably had serial numbers before March 1983, there seems not to be a good reason for failing to keep track of them in the past except inadequacies in PSE&G management.

Q. Please discuss the control of maintenance work orders.

A. Some maintenance work orders were inadequately classified, controlled and reviewed and some completed work was inadequately inspected. A detailed review is given by PSE&G in Attachment F to their letter dated April 22, 1983, to the NRC. The NRC reviewed this material in NUREG 0977, page 6-4 and Appendix F. (Exhibit SHH-21, page 25) This review was summarized in the NRC Commissioner Briefing on April 26. The problems and the NRC-directed PSE&G review of some 16,000 maintenance work orders are summarized on pages 12 through 21 of that transcript. (Exhibit SHH-11, pages 2-11) The objective was to see whether there were any other serious lapses aside from maintenance work orders for the reactor trip breakers, which were incorrectly classified Non-Safety Related. Some 35 additional work orders were found incorrectly classified, and 34 maintenance work orders had inadequate documentation. These were each followed up and the NRC Staff concluded that there were no other serious safety lapses.

Q. What is the significance of these misclassified maintenance work orders?

A. Misclassifying this work as non-safety related meant that it got neither the engineering check nor the quality assurance that would have been performed for properly classified safety related work.

Q. Just exactly which work are we talking about?

A. The notice of violation lists the maintenance work orders which were not adequately controlled, which had a direct effect on the events of February 22 and 25, 1983. In particular, Work Order 925774, issued to perform the work on Unit 1 reactor trip breakers in January 1983, was incorrectly classified as non-safety-related. (Exhibit SHH-6, page 8) The chances of catching the mistakes would have been improved if the work order and the quality assurance review before and after work had been completed had been specified by correctly classifying the maintenance work order.

Q. Earlier you referred to in-service testing and stated that there were deficiencies. Please discuss this further.

A. The testing of the reactor trip breakers was not done adequately, as required by NRC regulations and Salem license conditions. (Code of Federal Regulations, Title 10, Part 50, Appendix B, Paragraph XI; Exhibit SHH-23)

this test. An operative shunt trip attachment will trip the circuit breaker and failure of the undervoltage trip attachment will be unnoticed. Page 12 and 13 of Exhibit SHH-12 list the post maintenance operability tests performed after the breaker failures on August 20, 1982, and January 6, 1983. As can be seen, some of the tests were performed only with the manual switch, the ability of some breakers to be tripped via their undervoltage trip attachments was not adequately tested.

Q. What is wrong with the manual test?

A. There are two different ways to trip one of these circuit breakers and disconnect the electrical circuit which it controls. The reactor protection system automatic reactor trip involves the action of the undervoltage trip attachment, which is normally energized during reactor operation. De-energizing an undervoltage trip attachment by cutting off the power to its coil causes it to trip the breaker and open the circuit which the breaker controls. That cuts off the power to all the rod drive mechanisms and shuts down the chain reaction, as previously described.

In addition to the undervoltage trip attachment, each reactor trip breaker has a shunt trip attachment. This is another component of a circuit breaker assembly. The shunt trip attachment works just oppositely to the undervoltage trip attachment. During normal operation the shunt trip attachment is quiescent and not connected to any electrical supply. If a source of electrical power should be connected to the shunt trip attachment, energizing it electrically, it actuates and trips the breaker, disconnecting the power between the power source and the rod drive mechanisms. It can be seen that the shunt trip attachment, which trips the circuit breaker when it is energized, works just in reverse to the undervoltage trip attachment, which is normally energized and trips the circuit breaker when it is de-energized.

Q. How are the shunt trip attachment and the undervoltage trip attachment used in the reactor trip system at Salem?

A. In the Salem design, only the undervoltage trip attachments are connected to the reactor protection system. Only the undervoltage trip attachments are ever involved in accomplishing the automatic safety trip of all the control rods, as carried out by the reactor protection system. Therefore, the undervoltage trip attachments must be tested to verify the operability of the automatic reactor trip, and this is not tested when the manual switch is used.

The NRC staff summarized by stating:

SECTION 5. OPERATIONS AND TRAINING

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. This section of my testimony, together with the following three sections (thus, Sections 5, 6, 7, and 8) consider additional aspects of PSE&G mismanagement as disclosed by the February events and the subsequent investigation. These sections go beyond the inadequate and incorrect maintenance and testing which was the direct cause of the failures.

In this section, I discuss PSE&G mismanagement of operations and training that contributed significantly to the outage.

The operations staff had been inadequately trained and did not adequately understand some important control-room information. This led them to miss identifying the February 22 failure until 4 days later, even though they had enough information to do so at the time. This serious deficiency in PSE&G management procedures was shown by this event; they were not adequate to ensure the required systematic review of abnormal events.

Q. Please discuss the operational staff response to the incidence of February 22 and 25.

A. The NRC concluded that while they shut the plant down safely in both events, the control room operating staff and their supervisors on shift at the time of the events did not fully understand what was going on. Some members of the staff lacked familiarity with some of the instruments.

The NRC found:

> They did not appear to understand whether specific first-out annunciator signals resulted from a trip demand or if the annunciators were confirmatory indicators... Similarly, the operators did not know what condition would cause a real reactor trip indication on the IRP4 reactor protection system control panel.
>
> (Exhibit SHH-21, page 16)

This same information is summarized in the Restart Safety Evaluation Report. (Exhibit SHH-9, page 8)

The NRC investigators concluded that a previous history of false alarms and other kinds of failures had resulted in the operators not fully trusting the information they see. One NRC investigator suggested that the PSE&G operators may have been correct in not trusting the first-out panel. (NRC

February 25. Therefore, no repair was perceived to be needed after the February 22 event and the system remained inoperative.

Q. Why didn't PSE&G perceive the failure on February 22?

A. Because the review immediately after the event of February 22 concentrated on the plant transient and the various equipment trips and outages that occurred. An operator cleared the first-out panel without noticing which indicator on that panel was lit. (Exhibit SHH-9, page 12) Apparently it is not common practice to observe and log the indication on the first-out panel even though that will tell you which signal and which equipment was suppose to have caused the reactor shutdown. (Exhibit SHH-2, pages 2-3)

Observation and a bit of analysis of the first-out panel indicators would likely have given a clue to the failure of the automatic reactor trip. It was wrong to clear the panel without reading and recording its indications.

The definitive information that was available on the failure is contained in the printout of the sequence of events recorder. PSE&G has stated that these data were misinterpreted by station personnel. (Johnson testimony, page 3) The NRC reviewers found:

> The licensee did not determine that there had been a failure to trip automatically on February 22 until the computer printout of the sequence of events was reevaluated on February 26, as a result of NRC inquiries.
>
> (Exhibit SHH-9, page 18)

The NRC reviewers concluded that very few people at the plant understand the sequence of events recorder. (Exhibit SHH-9, page 20) Mr. Starostecki of the NRC stated:

> I have to say this plant is fortunate to have a plant computer that prints out this kind of information in real time. There was no procedure telling the people how to do that evaluation, there are some basic questions that aren't answered.
>
> The operators, they don't know whether that signal originated by demand signal, the same problem we had with the annunciator, whether it is a confirmatory signal or a demand signal.
>
> (Exhibit SHH-2, page 13)

Q. What was the result?

A. The plant was restarted with the reactor trip system in a failed, inoperable condition and operated that way until the failure on February 25 resulted in the prolonged shutdown which is the subject of this testimony. The NRC staff concluded that Public Service Electric and Gas Company's failure to identify the inoperable condition of the reactor trip system was a violation of the plant technical specifications deserving of a large civil penalty.

Q. Is this an isolated instance of inadequate review by PSE&G?

A. No. The NRC reviewers found that:

> At the time of the February 22 event, PSE&G did not have a requirement which would ensure a thorough and systematic evaluation of reactor trip events. Although there was a written procedure, it required only that the cause of a reactor trip be determined before restart and identified the personnel who could authorize plant restart after a trip. The procedure stated that if the cause of the reactor trip had been identified and corrected, the operating engineer could make the decision for restart, but if the cause for reactor trip could not be identified, higher management had to authorize the restart. There was no requirement for additional review of plant trips by other personnel; e.g., the Site Operations Review Committee. The Shift Technical Advisor (STA) was directed to conduct an investigation of all incidents, complete the initiation section of the incident report, and determine the reporting requirements of the event. (Exhibit SHH-20, page 22)

These same reviewers stated:

> The basic cause for the licensee failing to identify that an ATWS event had occurred on February 22, 1983 was the lack of thorough and systematic review necessary to achieve a complete understanding of the event.
>
> (Ibid.)

SECTION 6. PROCUREMENT

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. In this section I give the evidence for PSE&G mismanagement of procurement activities for reactor trip breakers, which contributed to the failure and thus to the outage. After an investigation of these and other procurement activities of PSE&G, the NRC reviewers concluded that the PSE&G procurement procedures were in principle acceptable if rigorously implemented and enforced. However, the rigorous implementation and enforcement of the procedures was not adequately pursued by PSE&G and substantial errors resulted. (Exhibit SHH-6, page 1; and Exhibit SHH-9, page 15)

Q. Why is this significant?

A. Incorrect procurement documents can result in the receipt of unsuitable or incorrect equipment.

Q. Were the technical documents that went with procurement of the reactor breakers adequate?

A. While the original documents were adequate, incorrect documents were used for purchasing replacement breakers and replacement parts including undervoltage trip attachments.

The purchase specification for the original and replacement circuit breaker undervoltage trip attachments has the wrong voltage rating specified for the coil on the undervoltage trip attachments. (Exhibit SHH-21, page 20)

The purchase documents do not include the design modifications of Westinghouse document NCD-Elec-18, discussed earlier in Section 4 of my testimony.

Spare reactor trip breakers and spare parts for them including undervoltage trip attachments were purchased using documents that were misclassified in various ways besides the technical errors I discussed just above. The NRC reviewers evaluated many procurement document packages. In Exhibit SHH-21, pages 32-4, they discuss in some detail problems with 15 of these packages. These problems include incorrect seismic classification, lack of review by the quality assurance and sponsoring engineers, incorrect part numbers and other identification.

Q. Please continue.

A. The review by Franklin Institute showed that acceptance criteria were needed for parameters affecting the correct operation of the undervoltage trip attachment, together with testing methodology to be established for acceptance testing. Life testing was also needed as well as criteria

working plant documents for the procurement and maintenance of the reactor trip breakers and their components nor the parts list were classified as safety related.

The Master Equipment List, which provides a detailed list of the equipment down to the component level and is used as a reference document for determining the safety classification of equipment, had several problems. It did not include the reactor trip breakers or components even though they are safety equipment. (Exhibit SHH-6, page 6) Procedures for the use of the master equipment list were inadequate. The distribution of the master equipment list was not controlled. The list was not being used by station personnel since they considered it to be incomplete. (Exhibit SHH-6, page 6)

Q. Why is this important?

A. Misclassifying a purchase of safety equipment as non-safety means that the documents and the equipment won't get the special handling or the engineering and quality assurance reviews that safety equipment receives. Any mistakes would be more likely to be identified and corrected by the extra checks that go with the safety classification.

Q. Dr. Hanauer, what do you conclude?

A. I conclude that PSE&G's procurement program was inadequately managed; that it allowed the use of incorrect and obsolete technical documentation, and that the incorrect non-safety classification allowed these errors to pass without engineering review or quality assurance. The result included acquisition of equipment that apparently did not conform to the correct design, plus the necessity, after the February 1983 events, for extensive re-review of past purchases that contributed to an unknown degree to the length of the outage.

SECTION 7. PSE&G MANAGEMENT OF THE WORK OF CONTRACTORS

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. The purpose of my testimony here is to document and evaluate how PSE&G managed the activity of their contractors.

Q. Why were contractors used?

A. As for all power plants, the equipment installed at Salem was purchased from vendors of such equipment, or was designed and constructed by organizations serving as PSE&G contractors or subcontractors. PSE&G has stated that errors of omission and commission by Westinghouse are involved in the reactor trip breaker failures at Salem and that PSE&G

(2) inadequate attention by the licensee [PSE&G] to the importance of vendor-supplied information, including management control of vendor-supplied information

(3) absence of an adequate preventive maintenance program, which could have highlighted and initiated the resolution of this problem and perhaps many others.

Root causes of these problems are considered to be failings in the following features necessary to assure adequate plant reliability though receipt and maintenance of vendor information:

(1) management policy at both PSE&G and Westinghouse requiring reliable distribution, maintenance, and verification of currency of vendor information

(2) procedures within both Westinghouse and PSE&G for generating, distributing, and maintaining vendor information

(3) training of personnel in these procedures and in the significance of adherence to them

(4) management controls to assure adherence to the procedures

(Exhibit SHH-20, pages 26-7)

Q. Please discuss Westinghouse errors in design and PSE&G management of them.

A. The Westinghouse errors related to their 1971 design changes have already been discussed in my testimony. PSE&G did not reference the changed specifications on procurement and maintenance documents for over ten years. Yet they now say they received NCD-Elec-18 in 1972. (Exhibit SHH-22, page 6) In my judgment, a more rigorous and inquisitive management of the activities of Westinghouse on the part of PSE&G would have brought these discrepancies to light and resulted in their correction.

Q. Please discuss the Westinghouse errors in maintenance information furnished by Westinghouse.

one occasion for maintenance and repair, but did not adequately ensure that the work performed and the information furnished were adequate. In fact, Westinghouse work and information were inadequate, and the lack of adequate management control by PSE&G led to failure to detect and correct the problem. The serious deficiencies in Westinghouse activities could have and should have been detected and corrected by PSE&G using practical management controls.

SECTION 8. QUALITY ASSURANCE

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. To discuss inadequacies in the PSE&G quality assurance program disclosed by the events of February 1983 as further manifestations of PSE&G mismanagement.

Q. What is the function of the quality assurance program at Salem?

A. The NRC reviewers state that at a nuclear power plant quality must be everybody's job and that the functions of quality assurance and quality control are to assure that jobs important to safety are done correctly. (Exhibit SHH-20, page 31)

Quality assurance should encompass more than the NRC "Important to Safety" purview. In a power plant, nuclear or otherwise, the function of quality assurance is associated with the efficiency and economy of operation as well as safety. It was PSE&G's obligation to develop and implement a quality assurance program at Salem that would in fact assure that jobs important to safety and economy were done correctly.

Q. What were the problems with the quality assurance program at Salem?

A. Much of my preceding testimony relates to failings in one or another aspect of the quality assurance program at Salem. Many, if not all, of the errors and failures that contributed significantly to the outage that started on February 25, 1983, could and should have been detected and rectified by an adequately functioning, alert and aggressive quality assurance program.

I will not reiterate again the specific inadequacies in the Salem programs that have already been discussed in my testimony.

Q. Was information available before February 1983?

SECTION 9. MANAGEMENT

Q. Dr. Hanauer, what is the purpose of this section of your testimony?

A. In this section of my testimony, I give some evaluations of the overall performance of PSE&G management of the Salem Nuclear Power Station, and conclude that, in addition to the specific program deficiencies that have been and are being remedied by the corrective action program, there is a more deep-seated inadequacy in PSE&G management which was a root cause of the failure and, therefore, a substantial contribution to the outage.

Q. What do you mean?

A. The NRC Restart Safety Evaluation Report puts it this way.

> During the fact-finding team review during the first week of March 1983 and concurrent analysis of the breaker failure events, licensee treatment of the reactor trip breakers and the circumstances surrounding their failure on February 22 and 25, provided the NRC staff with several indicators suggesting a major breakdown in management and quality assurance program implementation at the Salem Nuclear Generating Station. Subsequent detailed reviews and evaluations by the licensee and the NRC staff have confirmed that the programs in place are basically sound. Two aspects of these programs surfaced as the principal causes of the events discussed in this safety evaluation. The first of these was a perceived lack of resolve on the part of managers and supervisors in enforcing adherence to procedures by station personnel. The second aspect relates to the safety perspective displayed by corporate management in providing policy direction and priorities to the operating staff and the three existing review committees.
>
> (Exhibit SHH-9, pages 25-6)

Additionally these reviewers stated:

> Historically, however, PSE&G management has not displayed the expected aggressive effort to self evaluate and redirect efforts to correct internally

A. No. PSE&G Witness Morris states that safety must come first and NRC regulations must be met. Mr. Morris characterizes the NRC standards as requiring perfection and, since perfection cannot practically be achieved, deviations will occur and NRC penalties, including outages, will be incurred. (Morris testimony, pages 14-15) Mr. Morris states that the local regulatory criterion should be whether PSE&G performed reasonably under the circumstances, and he concludes that PSE&G did behave reasonably in connection with the February outage.

Q. What is your opinion?

A. My opinion is that Mr. Morris's conclusions for this outage do not follow as he states from the principles which he has enunciated. I believe that Public Service Electric and Gas Company did not behave reasonably and prudently in the ways I have discussed throughout my testimony, and that the outage is to a substantial degree the result of their unreasonable and imprudent conduct.

Q. Isn't this demanding perfection?

A. No. I don't believe it is, but it is demanding that the owner of a nuclear power plant exercise, reasonably and prudently, the kind of management required of the owner of a nuclear power plant. The NRC standard is well known and well understood. PSE&G knows and understands the NRC standard, or should do so. If events that cannot reasonably and prudently be provided for force expensive plant outages, then those are certainly part of the cost of generating energy at a nuclear power plant. I believe that the management of PSE&G, like the management of all such plants, is required to take cognizance of nuclear power plant safety. I read Mr. Morris's testimony to agree with me in this. However, I believe the Salem plant outage was required by safety considerations and could have been avoided by cost-effective, reasonable and prudent management actions. The actions and omissions of PSE&G management contributed substantially to the outage cause or duration.

Q. Didn't PSE&G rely legitimately and reasonably on Westinghouse and aren't the errors attributable to Westinghouse?

A. PSE&G did not adequately manage and direct the activities of its contractor, Westinghouse, and must therefore be charged with its contractor's errors. (Please see Section 7 of my testimony for my analysis.)

Q. What about the fact that the NRC has found that many of these problems are "generic"; that is, that other utilities have been discovered after the failures at Salem to have some of the same deficiencies as PSE&G.

If a combination of equipment failures and human errors did nevertheless cause a failure of the Reactor Protection System to function, this would almost surely have been identified by the operating staff of the plant.

And if such a failure were to occur, the consequent investigation into the proximate and ultimate causes would not have revealed the mismanagement that occurred at Salem, and so the extended outage would not have been required.

Q. What do you conclude?

A. I conclude that the mismanagement by PSE&G, in the circumstances of these events, is not just the inevitable shortfall from a standard of perfection, and is not excused by the fact that other utilities had some of the same problems.

SECTION 10. CONCLUSIONS AND RECOMMENDATIONS

Q. Dr. Hanauer, please give your overall conclusions from your evaluation.

A. I have concluded that:

1. The outage extension at Salem Unit 1, starting February 25, 1983, and lasting at least until May 8, was caused by PSE&G mismanagement resulting in two failures of the Reactor Protection System.

2. The failures of the Salem Unit 1 Reactor Protection System of February 22 and 25, 1983, revealed management deficiencies that induced the NRC to require the plant to remain shut down for an extended period. Restart was conditioned by the NRC on development by PSE&G of an extensive remedial program directed at the deficiencies, and accomplishment of all short-term tasks of this program.

3. The facts of the events are not in dispute. In both events, a signal from the safety instrumentation called for a reactor trip, but the control rods did not insert as required to shut off the neutron chain reaction in the reactor core. The breakdown was caused by concurrent failure of both Reactor Trip Breakers to open when called upon.

4. Deficiencies were revealed in the areas of operations, training, surveillance, maintenance, procurement, control of vendor activities and information, quality assurance.

State of New Jersey
Board of Public Utilities

Docket No. 831-25
List of Exhibits

Testimony of Stephen H. Hanauer
December 1, 1983

SHH-1. Qualifications statement, Stephen H. Hanauer

SHH-2. Transcript of NRC Commission Meeting 3/15/83, selected pages

SHH-3. Transcript of NRC Commission Meeting 3/24/83, selected pages

SHH-4. Letter Starostecki to Uderitz 3/7/83

SHH-5. Letter Eisenhut to Uderitz 4/29/83

SHH-6. Notice of Violation Letter & Enclosure 5/5/83

SHH-7. Order & Appendix Letter & Enclosure 9/29/83

SHH-8. INPO SOER 83-8, selected pages

SHH-9. Restart SER 4/28/83

SHH-10. Denton Restart Order 4/28/83

SHH-11. Transcript of NRC Commission Meeting 4/16/83, selected pages

SHH-12. 4/22/83 Letter Uderitz to NRC, Comments on NUREG 0977

SHH-13. Procedure M3Q-2 selected pages

SHH-14. INPO 1981, selected pages

SHH-15. INPO 1982, selected pages

SHH-16. PSEG Letter 4/22/83 responding to bulletin, selected pages

SHH-17. 7/6/83 PSEG Letter responding to Exhibit SHH-6, selected pages

SHH-18. PSEG NRB Meeting 83-05

SHH-19. PSEG SORC Meeting 83-02

SHH-20. NUREG-1000, selected pages

SHH-21. NUREG-0977, selected pages

SHH-22. Letter Selover to DeYoung 10/28/83

SHH-23. Code of Federal Regulations, Title 10, Part 50, Appendix B, Paragraph XI.