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U.S. House of Representatives

Committee on Energy and Commerce

SUBCOMMITTEE ON ENERGY AND POWER

Washington, DC 20515

March 16, 1987

Mr. David A. Ward, Chairman
Advisory Committee on Reactor Safeguards
1717 H Street
Washington, DC 20555

Dear Mr. Ward:

The Subcommittee on Energy and Power is investigating the implications for the safety of nuclear power plants of the recent Surry accident. In particular, we are concerned that (1) despite the designation of the failed feedwater line as "a nonsafety related system," a similar failure in a Boiling Water Reactor could result in the release of radioactive steam outside the containment structure; and (2) standards established for new nuclear power plants and inspection procedures for operational plants may not adequately take into account the possibility of deterioration of materials.

We are requesting your response to the following questions:

1. The NRC Augmented Inspection Team Reports Nos. 50-280/86-42 and 50-281/86-42 (NRC team reports) indicate that the failure at the Surry Station was caused by service induced deterioration of the feedwater suction line between the condenser and the feedwater pump.

   (a) What codes, standards, specifications and regulatory requirements are applied to the failed feedwater line and associated equipment (condenser, feedwater pumps, steam turbine, pipelines and components)? Are these systems classified as nuclear or non-nuclear? Are they classified as safety or nonsafety related systems?

   (b) Are these requirements different than those applicable to other portions of the feedwater and steam lines that are closer to the steam generators and reactor vessel? If so, why are they, and do you think this distinction is appropriate in view of what occurred in the Surry Plant accident? What is the safety justification for the differences?

(c) If a failure in the feedwater piping occurred at a similar location, e.g., between the condenser and feedwater piping in a Boiling Water Reactor nuclear power plant, could radioactive material be released outside the containment?

(i) If so, how much could be released and what would be the consequences to the surrounding area?

(ii) How are these areas of the feedwater and steam lines classified in Boiling Water Reactors?

(iii) In view of the Surry accident, do you think that the classifications of these areas of the power plant (including the steam turbine, condenser and feedwater pumps) are appropriate?

(d) What additional requirements could be applied to the feedwater lines, steam lines, steam turbine, feedwater pumps, condenser and related equipment to improve the safety of nuclear plant operation?

(e) Do you think the NRC should make any changes in its regulatory requirements for Surry or other nuclear power plants in order to implement lessons learned from the Surry accident?

2. The NRC team reports cited erosion/corrosion induced thinning of pipe metal as the cause of the failure at the Surry Station. Do the design, construction, maintenance or integrity monitoring codes, standards, or other regulations applied to nuclear power plants adequately provide for finding or make allowances for deterioration of plant components and piping in service? If not, what regulatory changes should the NRC make to incorporate these factors in plant design, inspection and maintenance requirements?

3. The two Surry Station nuclear units are very similar in design, nuclear reactor system and age. The units also "share" some support and auxiliary functions.

   (a) In view of this dependency, does it seem appropriate that Unit 1 was not shut down immediately when the failure occurred in Unit 2?

   (b) Should the NRC issue any new regulatory guidance for such situations?

4. Changes in the control room ventilation system were being implemented while the plant was running and at the time of the accident. The NRC inspection team reports conclude that the modification work resulted in the control room being flooded with potentially lethal carbon dioxide gas.

(a) Are NRC regulations adequate for modifications being performed while plants are operating? Were these regulations being observed at the time of the accident?

(b) Do you feel that different procedures should have been used? Should the NRC make any regulatory changes to prevent ongoing modification work from compromising operational safety?

5. The NRC inspection team reports indicate the accident was initiated by an improperly maintained valve.

(a) Does it seem appropriate that the plant was allowed to operate with this valve not functioning properly? Are there adequate requirements for inspections of such valves?

(b) Should the NRC make any regulatory changes as a result of the maintenance deficiencies discovered during the investigation of this accident?

6. What actions independent of NRC regulatory requirements should the industry take to implement lessons learned from the Surry accident?

Thank you for your assistance with this investigation. We would appreciate having your response no later than April 10.

Sincerely,

Philip R. Sharp

Philip R. Sharp
Chairman

PRS:bh

Jack H. Ferguson
President and
Chief Executive Officer

Post Office Box 26666
Richmond, Virginia 23261
804-771-4271

VIRGINIA POWER

April 9, 1987

The Honorable Philip R. Sharp
Chairman, Subcommittee on Energy and Power
Committee on Energy and Commerce
U. S. House of Representatives
Washington, D. C. 20515

Dear Representative Sharp:

On March 16, 1987, you informed us of your intent to investigate the implications of the December 9, 1986 Surry 2 feedwater pipe rupture. You requested that we assist you in that investigation by providing responses to six questions contained in your letter. Our responses are attached.

As indicated in my March 20, 1987 letter, we would be happy to discuss our responses with you or the Subcommittee staff in a meeting that would facilitate the most complete understanding of this information.

Very truly yours,

J. H. Ferguson

Attachment

cc: Mr. L. W. Zech, Chairman
U. S. Nuclear Regulatory Commission

Mr. W. H. Owen, Chairman
NUMARC Steering Committee

Mr. Z. T. Pate, President
Institute of Nuclear Power Operations

Mr. J. J. Taylor, Vice President
Electric Power Research Institute

Attachment

Question 1(a)

The NRC Augmented Inspection Team Reports Nos. 50-280/86-42 and 50-281/86-42 (NRC team reports) indicate that the failure at the Surry Station was caused by service induced deterioration of the feedwater suction line between the condenser and the feedwater pump.

What codes, standards, specifications and regulatory requirements are applied to the failed feedwater line and associated equipment (condenser, feedwater pumps, steam turbine, pipelines and components)? Are these systems classified as nuclear or non-nuclear? Are they classified as safety or nonsafety related systems?

Response

The codes, standards, and specifications to which the feedwater/condensate piping was designed and built are:

- United States of America Standard Code for Pressure Piping USAS B31.1.0 Power Piping, 1967 Edition, plus all applicable code cases
- ASME Boiler and Pressure Vessel Code
- ASTM Specifications
- Manufacturers Standardization Society of the Valve and Fitting Industry
- Section IX Welding Qualification of the ASME Boiler and Presssure Vessel Code
- American Welding Society Specifications
- Pipe Fabricators Institute

The equipment associated with the feedwater/condensate piping was designed and built to equipment manufacturers standards at the time of procurement (circa 1968). For example, the condenser and feedwater heaters were built to Heat Exchange Institute (HEI) standards. The feedwater heaters were also built in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

The systems associated with the failed feedwater/condensate piping are not classified as "nuclear" as defined by USAS B31.1.0 Code Case N1, and are considered conventional piping.

The condensate piping systems are classified as nonsafety-related except for the emergency condensate storage tanks and the piping systems from these tanks to the suction side of the auxiliary feedwater pumps. These components are classified as safety-related and are seismically supported.

The feedwater system piping is classified as nonsafety-related except for piping, valves, and supports from the steam generators to and including the first isolation (check) valve outside containment; auxiliary feedwater pumps; and the piping, valves, and supports from the auxiliary feedwater pumps to the main feedwater lines. These components are classified as safety-related and are seismically supported. The feedwater regulator valves are classified as safety-related but are not designated as seismically supported components.

Question 1(b)

Are these requirements different than those applicable to other portions of the feedwater and steam lines that are closer to the steam generators and reactor vessel? If so, why are they, and do you think this distinction is appropriate in view of what occurred in the Surry Plant accident? What is the safety justification for the differences?

Response

Yes, construction requirements for the safety-related portions of the feedwater and main steam lines were more stringent. The feedwater piping between the steam generators and the first isolation (check) valve outside containment and for the main steam piping from the steam generators to the non-return valves were subjected to additional inspections; i.e., all welds in these piping systems were 100% radiographed (x-rayed). These additional inspection requirements were established to insure weld integrity and supplement the verification of quality workmanship in implementing the piping system design.

Imposing the additional safety-related piping weld inspection requirements would not have prevented the piping rupture event at Surry Unit 2. The event was caused by a flow-induced erosion/corrosion phenomenon unrelated to the weld integrity of the piping. Even if current weld inspection criteria had been used in the design and construction of the feedwater/condensate piping, the erosion/corrosion phenomenon at Surry would not have been prevented.

The design criteria required by USAS B31.1.0 for calculating the piping minimum wall thickness (pressure boundary) and the materials used for the feedwater/condensate piping are identical for the safety and nonsafety-related portions of the piping.

Regarding the question on differing requirements for safety and nonsafety-related systems or components, the distinction is justified to assure that public health and safety is protected and that there is no undue risk from operation of a nuclear plant. The industry, and regulators, require very high standards of performance for those systems and components necessary for nuclear safety. We place special emphasis on the systems, components and structures needed to prevent or mitigate the consequences of postulated radiological accidents, and to shut down or maintain the unit in a safe shutdown condition. Nevertheless, portions of the plant not associated with nuclear safety, for example, power production or turbine support systems, are also held to high performance and industrial safety standards established within the electric utility industry.

Question 1(c)

If a failure in the feedwater piping occurred at a similar location, e.g., between the condenser and feedwater piping in a Boiling Water Reactor nuclear power plant, could radioactive material be released outside the containment?

(i) If so, how much could be released and what would be the consequences to the surrounding area?

(ii) How are these areas of the feedwater and steam lines classified in Boiling Water Reactors?

(iii) In view of the Surry accident, do you think that the classifications of these areas of the power plant (including the steam turbine, condenser and feedwater pumps) are appropriate?

Response

North Anna and Surry Power Stations use Westinghouse-design pressurized water reactors which Virginia Electric and Power Company (Virginia Power) is licensed by the NRC to operate. We are fully qualified to address questions regarding their design, construction and operation. However, we have no practical experience with boiling water reactors and thus do not consider ourselves qualified to respond to questions regarding such designs.

Question 1(d)

What additional requirements could be applied to the feedwater lines, steam lines, steam turbine, feedwater pumps, condenser and related equipment to improve the safety of nuclear plant operations?

Response

We have considered the question of "safety" from three perspectives: nuclear (radiological) safety, potential system interactions between safety-related and nonsafety-related systems, and finally, industrial (or non-radiological) safety.

From the nuclear safety perspective, no additional requirements should be applied. The regulatory requirements for periodic testing and inspection programs currently in place for safety-related systems provide adequate assurance that they will perform their intended safety functions. We also believe that the distinction between safety-related and nonsafety-related systems is appropriate for the reasons cited in response to Question 1.b.

The issue of system interaction in nuclear power plants is currently being examined by the NRC (designated as Unresolved Safety Issue A-17) in concert with industry groups and several nuclear utilities. The objective of this effort is to identify where the current design, analysis, and review procedures may not adequately account for potentially adverse systems interactions and to recommend action to rectify deficiencies. The current NRC position, pending the completion of this effort, is that existing regulatory requirements and procedures provide an adequate degree of public health and safety assurance.

As described in the NRC team report, certain system interactions did occur during the Surry event (i.e., inadvertent fire protection systems actuation, security system degradation). However, these interactions did not result in a reduction in nuclear safety. Proper operator/security force actions and the use of appropriate emergency systems (e.g., control room emergency ventilation) fully mitigated any system interaction effects.

Regarding industrial safety, we deeply regret the loss of four lives as a result of the Surry 2 accident. The activities currently underway within the industry (described in our response to Question 6) should assure that the lessons learned from the Surry 2 event are appropriately implemented at all power plants.

Although this event occurred at a nuclear plant, it was not a nuclear accident (i.e., involving radioactive materials) but rather an industrial accident. Other industrial facilities (e.g., industrial plants using heated, pressurized water or fossil-fuel power plants) could be susceptible to the erosion/corrosion phenomenon experienced at Surry.

On February 10, 1987, we conducted presentations across the country to disseminate information regarding the Surry 2 event. A number of major utilities with fossil-fuel plants attended. In addition, we are working with the Electric Power Research Institute (EPRI) and other industry groups to assure the broadest distribution and understanding of information related to the single phase liquid erosion/corrosion phenomenon.

Question 1(e)

Do you think the NRC should make any changes in its regulatory requirements for Surry or other nuclear power plants in order to implement lessons learned from the Surry accident?

Response

No. As nuclear industry groups address the Surry event, utilities will be receiving both the information and the technology necessary to correct the problem. No changes in regulatory requirements are necessary. The nuclear industry's ability to learn the lessons has improved significantly since the March 1979 accident at Three Mile Island. The creation of the Institute of Nuclear Power Operations (INPO) was the first of several steps toward that improvement. Part of INPO's mission is to "analyze events that occur in construction, testing, and operation of nuclear plants worldwide to identify possible precursors of more serious events; disseminate the lessons learned."

Utility groups, such as Nuclear Utility Management and Resources Committee (NUMARC), vendor owners groups, and industry groups such as the Electric Power Research Institute (EPRI), and the Atomic Industrial Forum (AIF) represent other mechanisms by which lessons learned have been shared. These groups are currently being folded under the umbrella of the Utility Nuclear Power Oversight Committee (UNPOC) to further improve industry's performance and enable it to work even more effectively with the Nuclear Regulatory Commission (NRC).

To that end, these industry organizations are being restructured into three broad areas: Regulation and Technical Support; Communication, Educational and Technical Services; and Government Affairs. The Regulation and Technical Support organization is intended to be the primary interface between the industry and NRC, although its scope will also include technical issues. This organization will encompass the functions of NUMARC - primarily the ability to present a unified industry position on issues. A NUMARC working group has been formed to address the erosion/corrosion phenomenon (see our response to Question 2).

Question 2

The NRC team reports cited erosion/corrosion induced thinning of pipe metal as the cause of the failure at the Surry Station. Do the design, construction, maintenance or integrity monitoring codes, standards, or other regulations applied to nuclear power plants adequately provide for finding or make allowances for deterioration of plant components and piping in service? If not, what regulatory changes should the NRC make to incorporate these factors in plant design, inspection and maintenance requirements?

Response

Yes, deterioration in service is considered. The original construction specifications applicable to this piping were in accordance with USAS B31.1.0. With respect to corrosion and erosion, USAS B31.1.0 states: "When corrosion or erosion is expected, an increase in wall thickness of the piping shall be provided over that required by other design requirements. This allowance in the judgement of the designer shall be consistent with the expected life of the piping." Our original design provided additional pipe wall thickness above that required for the internal system pressure which would have accounted for any expected corrosion. At that time, the complex phenomenon of erosion/corrosion was not generally recognized in the industry as a problem in single phase flow piping systems and therefore was not specifically evaluated. It is also important to recognize that piping systems made of stainless steel, or carbon steel containing low temperature, high oxygen water are not susceptible to this phenomenon.

In-service testing requirements for the safety-related portions of the systems are also imposed by the plant's Technical Specifications and Section XI of the ASME Boiler and Pressure Vessel Code for Inservice Inspection. In addition, Virginia Power is expanding its augmented program to include scheduled inspection, testing, and maintenance for applicable secondary-side piping.

Until the Surry pipe rupture event, the single phase liquid erosion/corrosion phenomenon was neither widely understood nor expected in power plant piping systems. However, the nuclear industry, in conjunction with EPRI, is developing a comprehensive understanding of the technical elements of erosion/corrosion. We can now discuss qualitatively the important variables affecting erosion/corrosion. Reliable nondestructive inspection procedures are available so that utilities can determine the extent of erosion/corrosion and measure its progression.

A NUMARC working group, chaired by Mr. W. L. Stewart, Vice President-Nuclear Operations, Virginia Power, is coordinating and evaluating these industry-wide inspection results. They will determine whether the scope of the concern justifies additional action by industry, and if so, what that action should be. We expect that this effort will identify factors in plant design, inspection, and maintenance requirements that may have to be modified.

Any regulatory change, should it be necessary, should only come as a result of a thorough examination of the benefits and liabilities associated with the change. We are confident that industry initiatives will more than satisfy the concerns of regulators and that no regulation to compel action will be required.

Question 3

The two Surry Station nuclear units are very similar in design, nuclear reactor system and age. The units also "share" some support and auxiliary functions.

(a) In view of this dependency, does it seem appropriate that Unit 1 was not shut down immediately when the failure occurred in Unit 2?

(b) Should the NRC issue any new regulatory guidance for such situations?

Response

3(a) Under the circumstances, it was appropriate that Unit 1 was not shut down immediately. Had Unit 1 been adversely affected, automatic safety systems as well as trained operations personnel were fully capable of shutting the unit down swiftly and safely. However, Unit 1 was judged by the onsite management and operations staff to be in a safe and stable steady-state operating condition and any precipitous action was deemed unwarranted until the event was better understood. In fact, placing Unit 1 in a transient condition similar to the one in progress on Unit 2 could have increased risk.

During the evening and night of December 9, 1986 we placed emphasis on initiating a preliminary investigation of the Unit 2 event, establishing a quarantined area to preserve evidence, bringing in needed specialists, working with regulators and the media, and establishing a recovery/investigation organization. Access to the Unit 1 Turbine Building was restricted to preclude personnel injury in the event of a similar occurrence on the Unit 1 side.

On December 10, following preliminary inspections of the Unit 2 pipe rupture, metallurgists had determined that the probable cause of the pipe failure was thinning of the pipe wall from the inner surface. Because the Unit 1 feedwater piping design was similar, they recommended inspection of Unit 1 piping. Virginia Power management immediately decided to shut Unit 1 down to inspect the wall thickness of piping. Shutdown of Unit 1 on December 10 was initiated as soon as Unit 2 was in a cold shutdown condition and the full attention of station personnel could be focused on the orderly shutdown of the operating unit.

We believe that these actions were responsible, well-considered, and, considering the circumstances, timely. We believe that it was appropriate to delay the shutdown of Unit 1 until we understood the nature of the event that had occurred on Unit 2 and were assured that the shutdown could proceed in a controlled manner.

3(b) No new regulatory guidance is needed. Because each potential event is unique, it is difficult for us to envision regulatory guidance that would provide information on how to handle unique events such as the one that occurred at Surry. Rather, the operating license and technical specifications already provide adequate regulatory guidance by defining the envelope within which the unit can be safety operated. In addition, reliance should be placed, as it is now, on a defense-in-depth design philosophy, redundant safety systems, highly trained and motivated personnel, and knowledgeable, responsible management to assure that appropriate and responsible actions are taken.

Question 4

Changes in the control room ventilation system were being implemented while the plant was running and at the time of the accident. The NRC inspection team reports conclude that the modification work resulted in the control room being flooded with potentially lethal carbon dioxide gas.

(a) Are NRC regulations adequate for modifications being performed while plants are operating? Were these regulations being observed at the time of the accident?

(b) Do you feel that different procedures should have been used? Should the NRC make any regulatory changes to prevent ongoing modification work from compromising operational safety?

Response

As described in the NRC's Augmented Inspection Team Report, 50-280/86-42 and 50-281/86-42, some carbon dioxide gas (CO_2) was present in the control room. However, the control room was not described as "flooded" with carbon dioxide. Rather, it experienced a mild ingress of CO_2/Halon. Personnel in the control room were able to carry out their operational duties safely. The NRC report attributed the CO_2 to the open doors into the control room area and discussed "modification" work on a ventilation fan as another possible source. The NRC reference was to a general area ventilation fan, 1-VS-AC-4, which is nonsafety-related equipment outside the control room area boundary. It supplies conditioned, fresh makeup air to several areas including the control room and is isolable by redundant, safety-related, motor-operated dampers. At the time of the accident, 1-VS-AC-4 was removed from service due to maintenance work (not modifications) and the isolation dampers were operable.

The control room has separate redundant safety-related systems for emergency air supply and filtration which are described in the Updated Final Safety Analysis Report (UFSAR) for Surry Power Station. The control room personnel turned on the emergency supply fans for the Main Control Room to disperse and dilute the CO_2, prevent its further infiltration, and supply fresh air to the control room. Additionally, two bottled air supply subsystems were available and ready for use in conjunction with the isolation dampers had it been deemed necessary. No modifications were being made to control room ventilation systems at the time of the accident; they were fully operable at the time of the accident. The ability to maintain a habitable control room environment under emergency situations was demonstrated.

NRC regulations governing modification activities are adequate and comprehensive. These regulations govern modifications to systems as described in the UFSAR. Developed to comply with NRC regulations, Virginia Power's design change program subjects system modifications to strict administrative controls with numerous safety, technical, management and independent organization reviews. In addition, modification and maintenance work on safety-related systems such as the control room emergency air supply systems is subject to strict operability requirements set forth in the Surry Power Station Technical Specifications.

Question 5(a)

The NRC inspection team reports indicated the accident was initiated by an improperly maintained valve.

Does it seem appropriate that the plant was allowed to operate with this valve not functioning properly? Are there adequate requirements for inspections of such valves?

Response

The deficiencies in the maintenance procedure did not affect the valve's ability to perform its intended safety function (i.e., to shut). Other administrative controls required that this capability be demonstrated successfully prior to returning the unit to operation. However, as noted in the NRC team report, the maintenance procedure used to overhaul the valve lacked detailed instructions, was not fully followed, and did not provide adequate documentation. These deficiencies have been corrected.

Current requirements assure that a quality maintenance program be established and implemented for safety-related valves. The main steam trip valve maintenance program is an ongoing program which provides adequate assurance that periodic inspection of these valves will be performed. The referenced maintenance deficiency applied to one particular aspect of one specific procedure and did not adversely affect the valve's ability to perform its intended safety function. We conclude that adequate requirements for valve inspections are already in place, that known deficiencies have been corrected, and that plant operation was appropriate because the valve's safety function had not been adversely affected.

We believe it is important to note that improper valve maintenance was not the cause of the Surry accident. Rather, the pipe rupture was the result of a chain of events: a normal pressure transient in the condensate system resulting from a reactor trip that caused the failure of a portion of piping that had been severely thinned due to erosion/corrosion.

Question 5(b)

Should the NRC make any regulatory changes as a result of the maintenance deficiencies discovered during the investigation of this accident?

Response

Current regulations require that administrative controls be in place to assure that maintenance activities are performed in a quality manner. The maintenance deficiencies that occurred at Surry were not as a result of any programmatic breakdown, but rather in our implementation of a specific maintenance procedure. We don't believe that any regulatory changes are necessary as a result of this single, isolated occurrence.

In response to concerns from both regulators and the nuclear industry about maintenance performance, a NUMARC Working Group was established in late 1984. Its objective was to facilitate and accelerate industry-wide maintenance improvement, assist with technology transfer, and improve the confidence that U.S. power stations are being properly maintained. An industry assessment of maintenance programs has been completed. Peer evaluations are underway. Event analyses have been conducted to determine the influence of maintenance on plant significant events. The Working Group has assisted INPO in upgrading evaluation criteria, developing a maintenance guideline document and installing a maintenance trend indicator program. The Working Group has interfaced with the NRC staff and with Standards committees in the maintenance area. These, and other industry efforts, are expected to continue under the reorganized industry groups (see response to Question 1.e.).

Question 6

What actions independent of NRC regulatory requirements should the industry take to implement lessons learned from the Surry accident?

Response

Since the event at Surry station, we have responded fully to every good faith inquiry related to it. We have sponsored industry seminars throughout the country to provide the widest possible dissemination of information about the phenomenon that led to the pipe rupture. In addition, we have worked closely with industry groups to make them aware of the possibility of piping deterioration. We have cooperated closely with INPO in issuing a Significant Event Report and a Significant Operating Experience Report. We have also helped establish a cooperative program at EPRI and a NUMARC working group to develop a unified industry position and determine appropriate action in response to the Surry event.

We believe that these actions, rather than any regulatory requirements, will be the most effective means of implementing the lessons learned from the Surry event.