UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

PREPARED DIRECT TESTIMONY OF DALE G. BRIDENBAUGH AND GREGORY C. MINOR ON BEHALF OF SUFFOLK COUNTY

REGARDING

SUFFOLK COUNTY CONTENTION 11

PASSIVE MECHANICAL VALVE FAILURE

April 13, 1982

SUMMARY OUTLINE OF SUFFOLK COUNTY
CONTENTION 11 TESTIMONY */

Suffolk County contends that LILCO performed an incomplete and inadequate analysis of potential unsafe conditions which would be caused by passive mechanical valve failures at Shoreham.

As documented in the Staff's SER, LILCO has only undertaken a limited review of potential passive mechanical valve failures. This review has omitted consideration of the types of serious main steam line isolation valve separation failures which have occurred at several operating U.S. BWR's.

The General Design Criteria require an Applicant to analyze potentially unsafe conditions affecting safety-related systems. LILCO's analysis is incomplete and thus does not comply with the requirements of 10 C.F.R. Part 50, Appendix A.

Exhibits */

1. Task No. B-58 from NUREG-0471
2. U.S. NRC Power Reactor Events, July - August 1981, Cover and pp. 7-10.
3. FSAR Section 6.3.3.3 and Tables 6.3.3-3 and 6.3.3-4
4. LILCO internal response to I&E Information Notice 81-28**/

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*/ ASLB Memorandum and Order, March 15, 1982, p.29

**/ In response to Suffolk County's discovery requests, LILCO has promised to provide Suffolk County with a copy of this document. Suffolk County has not yet received it, however, and therefore a copy is not attached hereto. After Suffolk County has received this document and determined that it will be moved into evidence in support of its direct testimony, a copy will be served on the Board and all parties.

PREPARED DIRECT TESTIMONY OF DALE G. BRIDENBAUGH AND GREGORY C. MINOR REGARDING SUFFOLK COUNTY CONTENTION 11

PASSIVE MECHANICAL VALVE FAILURE

I. INTRODUCTION

1. This testimony was jointly prepared and edited by Dale G. Bridenbaugh and Gregory C. Minor. A statement of the qualifications of Messrs. Bridenbaugh and Minor has been separately provided to this Board.

II. STATEMENT OF CONTENTION

2. The purpose of this testimony is to address Suffolk County Contention 11 as admitted by the Board as follows:

> Suffolk County contends that LILCO has not demonstrated that the valves used in the safety-related systems at Shoreham will not fail in an undetectable or unsafe mode, thereby jeopardizing the safe operation of Shoreham and violating 10 CFR 50, Appendix A, GDC 23, 34, 35, 37 and 40.

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

III. DISCUSSION OF ISSUES

III.A.: Background and Summary of Position

3. The essence of Contention 11 is that mechanical valves used in safety-related systems at Shoreham may fail in a mode that could be undetectable (i.e., an undetected failure of a passive or active component), thereby rendering a safety-related system inoperable without such inoperable status being known to the plant operators. A related but different concern is that the

valve failure could itself cause either an operator action or transient event which could adversely affect the safe operation of the plant.

4. An example of the first concern described in the preceding paragraph would be the separation of a valve disc from its stem in an instance where the valve is normally closed. If the valve then were called upon to open for the admission (for instance) of core cooling water, the valve would remain closed and preclude passage of cooling water, even though the valve operator itself would remain fully operable. Similarly, failure modes are possible whereby a normally open valve can fail but remain open and be non-closeable when the accident condition subsequently occurs.

5. An example of a passive valve failure causing a transient would be a case where a normally open valve, such as a Main Steam Isolation Valve (MSIV), experiences a disc-stem separation and closes without benefit of an outside signal. Such an event would disrupt steam flow and could cause a reactor scram.

6. The issue of concern in all such events is whether the effects of "single failures" in the valves have been adequately addressed in the design and review of the Shoreham safety-related systems. The general requirements that must be met are contained in 10 CFR 50, Appendix A, General Design Criteria. The definitions and explanations related to such events are contained in the Appendix A Definitions and Explanations which state:

> Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single

> occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.2/
>
> 2/ Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.1/

7. Section 6.3.1 of the Safety Evaluation Report (SER) indicates that only a limited evaluation of passive valve failures has in fact been performed at Shoreham. It states:

> The emergency core cooling system must retain its capability to cool the core in the event of a limited passive failure during the long-term recirculation cooling phase following a loss-of-coolant accident. The applicant has evaluated the effects on the emergency core cooling system of limited passive failures such as pump seals, valve seals, and measurement devices.2/

Even though ECCS systems are evaluated for single failures,3/ the passive component failures have the added problem of not being detectable without operator interaction. Thus, the discovery of passive failures could be delayed or, they may not be discovered at all. In the latter event, the operator may take incorrect action with regard to the system involved.

1/ 10 CFR 50, Appendix A.

2/ NUREG-0460, page 6-45 (emphasis supplied).

3/ FSAR, page 6.3-3.

8. The limited review described in the SER, taken in conjunction with the fact that the single failure design conditions (as identified in the GDC definitions and explanations) are still under development, leads us to the conclusion that the Shoreham safety-related systems have not been demonstrated to be in full compliance with the appropriate regulations.

III.B.: Industry Experience

9. Passive mechanical valve failures have been identified as a potential problem in the nuclear industry for some time. It was officially designated as a "B" task in the NRC's Program for the Resolution of Generic Issues Related to Nuclear Power Plants, published in NUREG-0410, January 1, 1978. Category B tasks were identified in that document as: "those generic technical activities judged by the Staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the Staff perceives a lesser safety, safeguards, or environmental significance than Category A matters."

10. Task No. B-58 was identified in that document as passive mechanical failures but further definition of the task was not specified. A subsequent NRC report, Generic Task Problem Descriptions, NUREG-0471, published in June, 1978, provided further definition of the Category B, C and D tasks. In this report, the scope of the passive mechanical failure work was identified: "This task involves a review of valve failure data in a more systematic manner to confirm the Staff's present

judgment regarding the likelihood of passive mechanical valve failures, categorize these and other valve failures as to expected frequency, specify acceptance criteria and determine if and how the results of this effort should be applied in licensing reviews."[4/] A copy of the title page of this report and the page on which Task B-58 is described is appended as Exhibit 1.

11. Based upon our discussions, we understand that there has been little progress at the NRC in the investigation of Task B-58 beyond the NRC's stated intent to review valve failure data as described in NUREG-0471. During the informal discovery meetings held between LILCO, Suffolk County, and the NRC staff over the past two years, the subject of passive mechanical valve failures has been discussed several times. Suffolk County consultants requested the NRC Staff to provide details of the status of its review on this subject. All that could be established as being available was a computer printout consisting of valve failures available through the Licensee Event Report system. A request of the NRC for further information on the nature of the review implied by the B-58 task description produced no additional information. We have similarly requested relevant information concerning industry experience from LILCO and have received no data.

4/ NUREG-0471, June 1978, p. B-76.

12. Also during informal discovery meetings over the past two years, we have pointed out, to both LILCO and the NRC Staff, that there are data available on passive mechanical valve failure experience at U.S. reactors and that these data should be evaluated for applicability (both as to valve manufacturer and mode of failure) to the Shoreham system. An example of relevant failure experience is contained in the U.S. NRC Power Reactor Events, July-August 1981, which reports on seven mechanical failures of main steam isolation valves occurring at Brunswick Unit 2, Brunswick Unit 1, and Hatch Unit 2. The failures which occurred involved the mechanical separation of the valve internals permitting the main disc to be separated from the valve stem. The first reported failure occurred at Brunswick Unit 2 in January 1976, and the most recent one reported was July 18, 1981, also at Brunswick Unit 2. A copy of the referenced Power Reactor Events report is appended as Exhibit 2. These valve failures raise concerns for the capability of valves to perform safety functions and for the increased challenges to safety systems -- challenges which all experts agree are undesirable. However, in response to Suffolk County Interrogatories, LILCO stated that it has performed no evaluations or analyses to determine if failures similar to those experienced at Brunswick Units 1 and 2 and Hatch Unit 2 are possible at Shoreham.[5/]

5/ Response of Long Island Lighting Company to Suffolk County Interrogatories and to Suffolk County Second Set of Interrogatories, March 16, 1982, p. 29.

13. The foregoing valve failures at other nuclear power reactors are particularly significant in this case. It appears that LILCO utilizes isolation valves from the same manufacturer and which are at least similar to the valves that have failed at other plants. In response to I&E Information Notice 81-28, LILCO has issued a purchase order for necessary replacement parts for Rockwell MSLIV's with deliveries scheduled for May, 1982. See LILCO Response to Suffolk County Interrogatories, March 6, 1982, p. 29. However, from the same answer to the Suffolk County Interrogatories, it is clear that LILCO has not undertaken any analyses to ensure that the same kinds or similar kinds of failures would be detected if they occurred in replacement valves as well. It is our position that the mere replacement of valves without necessary procedures to detect passive valve failures is inadequate to ensure that these serious concerns are addressed.

14. Moreover, the primary significance of the reported passive mechanical valve failures is as an example of the type of failure addressed by this particular contention. In our opinion, it is particularly disturbing that seven failures have occurred in a valve of such importance as the main steam isolation valves and which presumably receives a great deal of attention during design review and verification. It is our concern that similar problems of passive mechanical valve failures may also exist in the valves used in safety-related core cooling systems.

III.C.: Shoreham Valve Review

15. The Shoreham FSAR discusses single failure valve analysis in Section 6.3.3.3 and identifies the extent of the analysis conducted. A review of this Section confirms the Staff's SER statement that the FSAR review was "limited."

16. In the FSAR, the primary consideration given to single failure in valves had to do with assuring that different divisions of power supplies are provided, and considering the valve failures resulting from loss of power or operator error. No mention is made of the possibility of passive component failures, such as separation of a valve disc from valve stem, nor is any discussion provided of whether indication is provided to operators of such failure modes. The FSAR text does refer to FSAR Table 6.3.3-3, ECCS Single Valve Failure Analysis (summary), but a review of this table indicates that a very minimal analysis was performed. A copy of FSAR Section 6.3.3.3 and Tables 6.3.3-3 and 6.3.3-4 is appended as Exhibit 3.

17. LILCO has indicated that it expects to detect passive valve failures by testing.[6] However, complete testing of passive components is usually done during outages, and would be difficult to conduct during plant operation. Thus, there would be long periods of time where failures could go undetected.

6/ Response of Long Island Lighting Company to Suffolk County Interrogatories and to Suffolk County Second Set of Interrogatories, March 26, 1982, p. 28.

IV. CONCLUSIONS

18. It is our opinion that LILCO has not adequately demonstrated the acceptability under the regulations of the passive mechanical valves utilized in safety-related systems at the Shoreham Plant. The regulations require that failure of such valves be considered. The regulations go on to state, however, that the methodology for conducting this review has yet to be developed.

19. The Applicant's review as identified in the FSAR is insufficient. This is confirmed by the Staff's assessment, contained in the SER, that only limited passive valve failures (pump seals, valve seals and measurement devices) have been analyzed. Although examples of passive valve failures have been reported in the industry literature, neither the Staff nor the Applicant has evaluated the relevance of these failures to Shoreham. The fact that the methodology is still under development provides no excuse for the inadequacy -- indeed lack -- of the review conducted to date.

Appendix A of 10 CFR 50 states as follows:

> "The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include: (1) Consideration of the need to design against single

failures of passive components in fluid systems important to safety."7/

20. This testimony concerns passive mechanical valve failures in safety-related systems. Under the NRC's regulations, such systems must be highly reliable. Such reliability cannot be assured where the Applicant has no systematic means to detect such failures. Accordingly, since LILCO has developed no reliable or systematic detection means, we conclude that LILCO has not demonstrated compliance with the General Design Criteria.

7/ 10 CFR 50, Appendix A, Introduction.

EXHIBIT 1

TASK NO. B-58 FROM

NUREG-0471

ATTACHMENT B

NUREG-0471

GENERIC TASK PROBLEM DESCRIPTIONS

Category B, C, and D Tasks

Manuscript Completed: June 1978
Date Published: June 1978

Program Support Branch
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Task No.	Title
B-58	Passive Mechanical Failures

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

This task involves a review of valve failure data in a more systematic manner to confirm the staff's present judgment regarding the likelihood of passive mechanical valve failures, categorize these and other valve failures as to expected frequency, specify acceptance criteria and determine if and how the results of this effort should be applied in licensing reviews.

EXHIBIT 2

U.S. NRC POWER REACTOR EVENTS

JULY-AUGUST 1981

ATTACHMENT C

POWER REACTOR EVENTS

United States Nuclear Regulatory Commission

July-August 1981/Vol. ~~3, No. 5~~

This document is a bi-monthly summary of selected events that have occurred at nuclear power plants. These events have been taken from public information sources, namely Licensee Event Reports and NRC Inspection Reports, and are or have been under review by the NRC. They are considered informative to persons interested in the nuclear industry and may concern safety-related matters, such as personnel errors and equipment malfunctions. Although most summaries are published shortly after the events have occurred, it is sometimes necessary to publish others several months later. This may be due, for instance, to generic problems noticeable only after an extended period of time, or because of lengthy resolution of certain concerns. All events are reported in the belief that open communication benefits all parties.

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Washington, DC 20555

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personally should remove the tags and cut in the manifold. He entered containment, removed the tags from the inlet and outlet valves, and attempted to open the inlet valve. Since the valve was difficult to move, the technician requested assistance from his supervisor who assigned a second technician to assist him. Neither technician noticed the open vent and drain valves. When the inlet valve was cracked open, steam blew out through the open vent and drain valves, burning the second technician's leg. The technicians left the area immediately and another technician in the area called the control room and reported a leak on the A loop RTD manifold.

Since the operators could not assess the amount of the leak, they started cooling down per technical specification requirements. An operations engineer wearing a heat protection suit and an air pack later managed to isolate the leak by closing the vent and drain valves on the manifold.

The IAE technicians had misunderstood the instructions of their supervisor, and had attempted to perform work for which they were neither trained nor qualified by station policy. They were not aware that a process pipe such as the cold leg RTD bypass manifold would have vent and drain valves that must be closed. IAE technicians are allowed to cycle, tag, and clear tags only on instrument and root valves; operations and chemistry personnel tag all other equipment. A more experienced technician would have realized that IAE technicians were not allowed to operate the RTD bypass manifold valves, and would have recognized the red tags belonging to operations. The IAE supervisor should have realized when the first technician called for assistance in opening the valves that a serious error was being made, and that safety hazards could be present.

Although the amount of the leak could not be determined, the capacity of the positive displacement pump was never seriously challenged even during the cooldown. All of the safety injection pumps were available to maintain pressure and volume of the reactor coolant system had they been needed. The fuel in the core at the time of the incident was new, with little fission product inventory. Thus, little radiobiological hazard existed.

The immediate corrective action was to isolate the leaks by closing the manifold vent and drain valves. IAE technicians will receive special instructions on which valves they are allowed to operate and the entire tagging procedure. IAE supervisors will be counseled to be more careful in giving instructions to less experienced technicians.[9]

FAILURE OF MAIN STEAM ISOLATION VALVES

The NRC has under review seven reports of mechanical failures of the Rockwell-Edward Flite Flow Stop Valve, a "Y" pattern globe valve made by Rockwell International and used for main steam isolation valves (MSIV) at some BWR facilities. Operating BWR facilities using this valve include Brunswick Units 1 and 2, Cooper, Duane Arnold, Fitzpatrick, Hatch Unit 2, and Vermont Yankee. Of the seven reported mechanical failures, five occurred at Brunswick Unit 2 and one each occurred at Brunswick Unit 1 and Hatch Unit 2.

The valve components that have failed are shown in Figure 1. The piston assembly is attached to the main disk (2) by thread engagement and then restrained from unwinding by pin (4). The stem disk (1) is also attached to the stem (6)

1. STEM DISK
2. MAIN DISK
3. ANTIROTATION PIN - STEM DISK TO STEM
4. ANTIROTATION PIN - MAIN DISK TO PISTON
5. STELLITED DISK GUIDES, 3 RIBS, 120° APART (OUT OF SHOWN VIEW)
6. STEM

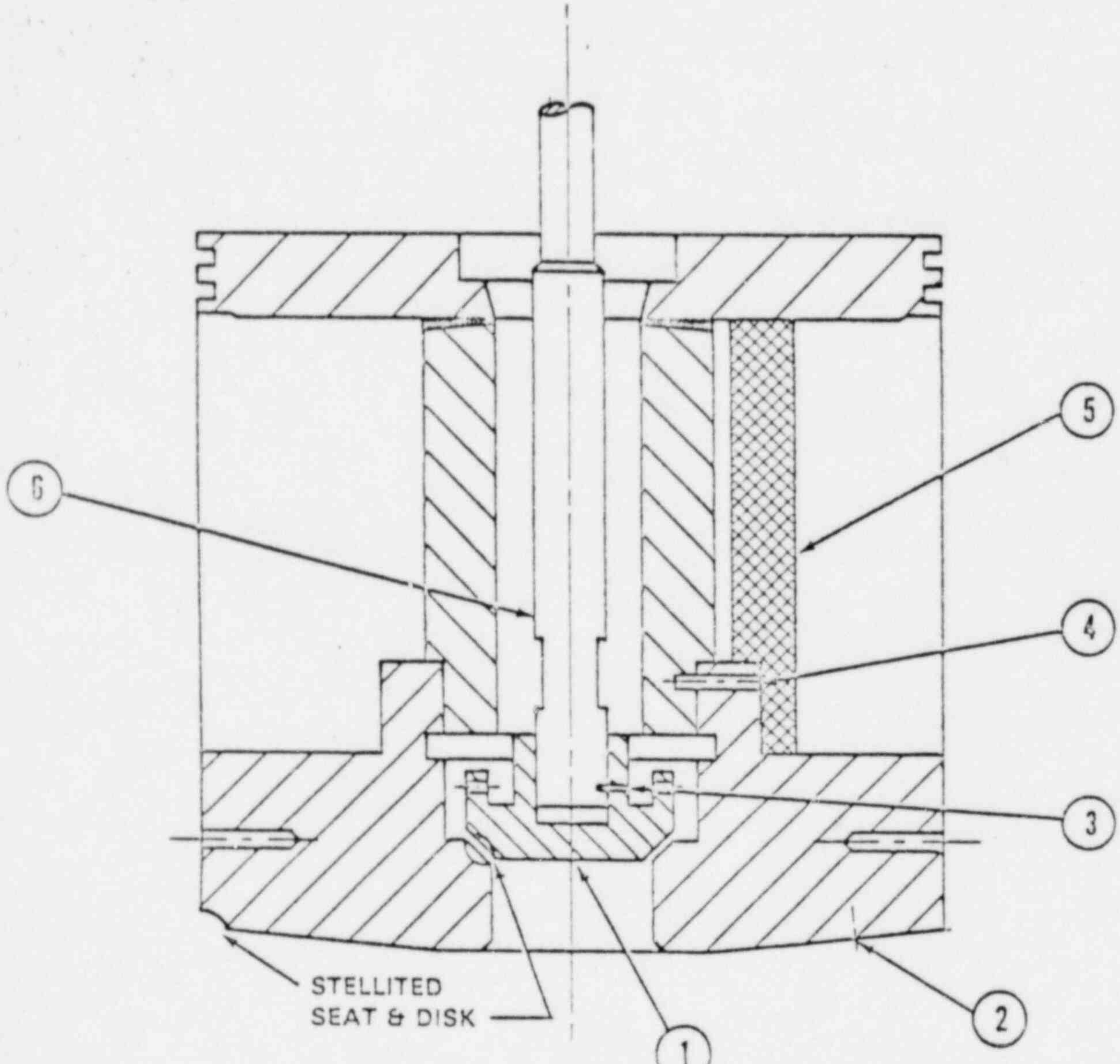

Figure 1. Main Disk Assembly of Flow Stop Valve

by thread engagement tightened to 1050 ft-lb torque specifications with an anti-rotation restraint provided by pin (3). The valve is closed primarily by spring forces. The valve is installed so that reactor steam pressure works to seat the main disk. The stem disk provides pressure equalization action to reduce overseat load for opening the main disk against system pressure.

The failures which occurred involved a mechanical separation of valve internals. This separation was either at the stem-to-stem disk threaded connection or at the main-disk-to-piston threaded connection. Either failure permits the main disk to be free of the stem. The failure in either mode results from failure of the threaded connection, which in some cases is caused by vibration-induced rotation of the disk so that it becomes disconnected from its threaded mate. Such rotation is believed to be caused by or aided by the propensity of steam flow to produce vibration and to create turning forces on valve internal components when anti-rotation restraint is inadequate due to a failed or missing pin. Other contributing causes are believed to result from reassembly of the valve after disassembly at the site; such reassembly may have included inadequately torqued connections and failure to properly install the pins. Also, an examination of spare parts at the Brunswick site showed that the thread dimensions on the stems and stem disks did not meet drawing tolerances.

The individual failures are described below in chronological order.

1. In January 1976 at Brunswick Unit 2, the D steam line inboard valve main disk separated from the piston. There was no evidence that pin (4) was installed.

2. On January 30, 1979 at Brunswick Unit 2, the A steam line inboard valve stem disk separated from the stem. A square pin was used in the round hole at point (3). The corners of the pin experienced high stress, thereby causing cracking of the pin.

3. On January 15, 1981 at Brunswick Unit 2, the C steam line outboard valve main disk separated from the piston. There was no evidence that pin (4) was ever inserted. This was deduced from finding the hole undamaged, with no sign of a plug weld.

4. On March 5, 1981 at Hatch Unit 2, the A steam line inboard valve stem disk separated from the stem. Pin (3) was not fully inserted.

5. On March 30, 1981 at Brunswick Unit 1, the C steam line outboard valve stem disk separated from the stem. Pin (3) failed and was not recovered.

6. On July 2, 1981 at Brunswick Unit 2, the C steam line inboard valve stem disk separated from the stem. Pin (3) was not properly installed.

7. On July 18, 1981 at Brunswick Unit 2, the D steam line inboard valve main disk separated from the piston. Pin (4) was not fully inserted.

These failures have raised concerns for the capability of the valve to perform its required safety function, and for the increased frequency of challenges to safety systems. However, it is noted that the failures to date have resulted in the main disk going closed (i.e., not cocking open) with some uncertainty only to its leak tightness.

Detailed investigation of the July failures at Brunswick Unit 2* led to the preliminary findings of possible excessive vibrations on valve internals from steam flow turbulences created by the piping direction changes. In addition, there was evidence of loose thread connections. Further investigations and evaluations are currently in progress. Preliminary corrective actions by the licensee include increasing the stem pin size from 5/16 inch to 3/8 inch, using three pins instead of one or two pins, and increasing the hole depth 1/8 inch into the stem. In addition, corrective actions for the main-disk-to-piston connection include adding an extra pin (of the same 1/2-inch size) and increasing the hole depth by 1/8 inch.

This matter remains under NRC review, and it is expected that licensees will review the information for applicability to their facilities.[10,11]

BROKEN THERMAL SHIELD BOLTS

On July 15, 1981, while conducting a 10-year inservice inspection program, the licensee at Oconee Unit 1** discovered loose parts in the bottom of the reactor vessel. These parts were discovered using a remote video camera, and were identified as parts from the thermal shield and its attachments.

The thermal shield is a 2-inch thick cylinder surrounding the core barrel; it extends the length of the core region. Its function is to provide additional shielding against gamma and neutron flux effects on the reactor vessel wall in the core region to reduce gamma heating in the reactor vessel wall and radiation effects on the vessel materials. The bottom support is shown in Figure 2.B. The inside diameter of the thermal shielding is machined to clear the bottom flange of the core barrel and to engage the lower grid with a diametral interference fit. Ninety-six 1-inch diameter, high-strength bolts secure the bottom end of the thermal shield to the lower grid plate. There were five missing bolts from this location.

The thermal shield's upper support (shown in Figure 2.A) consists of a stellite clamp and shim pad that are contoured to the thermal shield and core barrel curvature. Twenty of these assemblies are placed at equal intervals around the top end of the thermal shield and secured to the core barrel by high-strength bolts (three in each assembly). The design restrains the thermal shield radially both inward and outward, and allows axial motion to accommodate longitudinal differential thermal growth between the core barrel and the thermal shield.

Attached to the exterior of the lower internals are 12 pairs of lateral restraint guide blocks. Each half of the blocks is about 3" x 6.5" x 5" and

*An 821-MWe BWR located 3 miles north of Southport, North Carolina, and operated by Carolina Power and Light.

**An 887-MWe PWR located 30 miles west of Greenville, South Carolina, and operated by Duke Power Company.

EXHIBIT 3

FSAR SECTION 6.3.3.3

AND TABLES 6.3.3-3 AND 6.3.3-4

Criterion 3, Maximum Hydrogen Generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is stated in Table 6.3.3-2.

Criterion 4, Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 2, Section III, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

Criterion 5, Long Term Cooling

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in Reference 2, Section III.A. Briefly summarized, the core remains covered to at least the jet pump suction elevation, and the uncovered region is cooled by spray cooling and/or by steam generated in the covered part of the core.

6.3.3.2 Conclusions

Having shown compliance with the applicable acceptance criteria as summarized in Section 6.3.3.1, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10CFR50.46 acceptance criteria, given operation at or below the maximum average planar linear heat generation rates in Table 6.3.3-1.

6.3.3.3 Single Failure Considerations

The effects of a single failure or operator error that causes any manually controlled, electrically operated valve in the ECCS to move to a position that could adversely affect the ECCS has been studied. The purpose of this evaluation is to determine that any such malfunction does not affect the ECCS more than the results of the worst single failure which is reported in the LOCA calculations performed in accordance with 10CFR50 Appendix K.

The results of the break spectrum analysis show the single failure which results in the maximum calculated peak clad temperature (PCT). For any other single failure to be more significant, its effect on the ECCS must be greater than this single failure. Therefore, a study was made to determine if the malfunction of a manually controlled, electrically operated valve by some unknown cause or by an operator improperly positioning a control switch could affect the ECCS more severely than this failure.

In accordance with appropriate IEEE standards, the ECCS valves are electrically assigned to different divisions of power supply. The effect of an operator improperly actuating a single switch on the control panel is to cause only a single valve to move to an incorrect position. For the operator error of actuating a single switch of the ADS, the system valves are not actuated. However, the consequences of a malfunction which causes one ADS valve to inadvertently open has been noted.

The summary of the ECCS Valve Single Failure Analysis is provided in Table 6.3.3-3. Comparing the effects of the single valve failure noted in this table with the results of the Appendix K LOCA analysis, it can be seen that these failures are not more severe than those reported. The single failures considered for the ECCS analysis are presented in Table 6.3.3-4.

It is therefore only necessary to consider each of the above single failures in the emergency core cooling system performance analyses. For large breaks, failure of one of the LPCI injection valves is, in general, the most severe failure. For small breaks, HPCI failure is more severe than failure of a single ADS valve. ADS has no effect in large breaks. Consequently, as a matter of calculational convenience, it is assumed in all calculations that one ADS valve fails to operate in addition to the identified single failure. This assumption reduces the number of calculations required in the performance analysis and bounds the effects of one ADS valve failure and HPCI failure by themselves. The only effect of the assumed ADS valve failure on the calculations is a small increase (on the order of 100 F) in the calculated temperatures following small breaks.

An evaluation has been made of the potential for passive failures of fluid systems during long term cooling following a LOCA as well as single failure of active components. Single failure of passive components in electrical systems are assumed in designing against a single failure. ECCS are designed such that flow blockage of a single pipe cannot prevent a required safety output function. During the long term cooling mode the ECCS are operating at pressures and temperatures considerably below their design basis, thus the propensity for failure is reduced below that during operation at design conditions. Reactor pressure decreases with time and for the largest break area, which gives the highest PCT, the RHR system pressure decreases to less than 100 psi thus providing greater margin to the design basis and reducing the potential for passive failure.

A review has been conducted of the equipment arrangement of ECCS valves. The conclusion from this review is that all valves in the ECCS which must be operable during and after LOCA will not become submerged due to the occurrence of a LOCA. As a result, it is concluded that neither the short time requirements, nor long term cooling capability, is affected by submergence effects.

6.3.3.4 LOCA Analysis Procedures and Input Variables

The procedures approved for LOCA analysis conformance calculations are described in detail in Reference 2. These procedures were used in the calculations documented in paragraph 6.3.3.5. For convenience, the four computer codes are briefly described below. The interfaces between the codes are shown schematically on Figs. II-2a, II-2b, and II-2c in the "Documentation of Evaluation Models" Section II.A of Reference 2. For convenience, the major interfaces are briefly noted below.

Short-term Thermal Hydraulic Model (LAMB)

This code is used to analyze the short-term blowdown phenomena for large postulated pipe breaks (breaks in which nucleate boiling is lost before the water level drops and uncovers the active fuel) in jet pump reactors. The LAMB output (core flow as a function of time) is input to the SCAT code for calculation of blowdown heat transfer.

The LAMB results presented are:

1. Core average inlet flow rate (normalized to unity at the beginning of the accident) following a large break.

Transient Critical Power Model (SCAT)

This code completes the transient short-term thermal-hydraulic calculation for large breaks in jet pump reactors. The GEXL correlation is used to track the boiling transition in time and location. The post-critical heat flux heat transfer correlations are built into SCAT which calculates heat transfer coefficients for input to the core heatup code, CHASTE.

The SCAT results presented are:

1. Minimum critical power ratio (MCPR) following a large break.

2. Convective heat transfer coefficient following a large break.

Long Term Thermal Hydraulic Model and Refill/Reflood Model (SAFE/REFLOOD)

SAFE

This code is used primarily to track the vessel inventory and to model ECCS performance during the LOCA. The application of SAFE

TABLE 6.3.3-3

ECCS SINGLE VALVE FAILURE ANALYSIS

System	Valves (S)	Position for Normal Plant Operation Closed	Opened	Consequences of Valve Failure Assumed Together With Design Basis LOCA
Core Spray	Suction		X	Negate use of one core spray loop
	Injection (s)	X		Negate use of one core spray loop
	Test Return	X		Negate use of one core spray loop
	Minimum Flow		X	Partial loss of flow due to flow to the suppression pool
High Pressure Coolant Injection				
	Condensate Suction		X	Utilize suppression pool water
	Suppression Pool Suction Valve	X		Utilize condensate storage tank water
	Suppression Pool Test Return	X		Partial loss of flow due to flow to suppression pool
	Injection (s)	X	X	Negate HPCI
	Turbine Inlet (s)	X	X	Negate HPCI
Low Pressure Coolant Injection	Injection (s)	X	X	Negate use of LPCI
	Minimum Flow	X		Partial flow loss in one loop due to flow to suppression pool
	Test Return	X		No consequence
	Hx Bypass		X	Reduce flow due to Hx pressure drop
	Pump Suction		X	Negate one out of four pumps
Automatic Depressurization System	One Relief Valve	X		Vessel depressurizes faster, increases rate of HPCI injection (assuming the failure of a single ADS valve to open does not affect the results because the effect on small breaks is insignificant with HPCI in operation

TABLE 6.3.3-4

SINGLE FAILURE EVALUATION (WITH LPCI MODIFICATION)

The following table shows the single active failures considered in the ECCS performance evaluation.

Assumed Failure	Suction Break Systems Remaining	Discharge Break Systems Remaining
LPCI Injection Valve	All ADS, HPCI, 2 CS, 2 LPCI (1 loop)	All ADS, HPCI, 2 CS
HPCI	All ADS, 2 CS, 4 LPCI (2 loops)	All ADS, 2 CS, 2 LPCI (1 loop)
Diesel Generator A or B	All ADS, HPCI, 1 CS, 3 LPCI (2 loops)	All ADS, HPCI, 1 CS, 1 LPCI (1 loop)
Diesel Generator C	All ADS, HPCI, 2 CS, 2 LPCI (2 loops)	All ADS, HPCI, 2 CS, 1 LPCI (1 loop)
One ADS Valve	All ADS minus one, HPCI, 2 CS, 4 LPCI (2 loops)	All ADS minus one, HPCI, 2 CS, 2 LPCI (1 loop)
DC Source Common to HPCI and One Diesel Generator	All ADS, 1CS, 1LPCI (1 Loop)	All ADS, 1 CS, 1 LPCI (1 loop)

NOTE:

Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above designated failures.