PDR-016

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 23 1984

Ms. Nina Bell
Nuclear Safety Analyst
Nuclear Information and Resource Service
1346 Connecticut Avenue, NW
Washington, DC 20036

IN RESPONSE REFER
TO FOIA-84-275

Dear Ms. Bell:

This is in further response to your letter dated April 10, 1984, in which you requested two separate categories of documents relating to:

1. The TDI diesel generators at the Shearon Harris nuclear plant; and

2. All lists of problems and defects which have occurred with TDI generators being used or tested, or which have not yet been used for nuclear facilities and in other applications (e.g. marine).

Appendix A lists four additional documents relating to your letter. These documents are being placed in the PDR in file folder 84-275.

The search for remaining documents subject to your letter is still ongoing. You will be notified when our search is completed.

Sincerely,

John Philips for

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosure: Appendix A

APPENDIX A

1. 1/25/84 SECY-84-34 - Re: EMERGENCY DIESEL GENERATORS MANUFACTURED BY TRANSAMERICA DELAVAL, INC.
PDR/SECY/84-034 (4 pages)

2. 1/25/84 Memo for C. J. Heltemes, Jr., from H. R. Denton re: ABNORMAL OCCURRENCE REPORT TO CONGRESS FOR FOURTH QUARTER CY 1983 w/enclosure (24 pages)

3. 2/1/84 Letter to H. R. Denton re: TDI - EMERGENCY DIESEL GENERATORS w/attachment (13 pages)

4. 3/22/84 Memo for C. J. Heltemes, Jr., from H. R. Denton re: ABNORMAL OCCURRENCE REPORT TO CONGRESS FOR FOURTH QUARTER CY 1983 w/enclosed Rewrite of TDI Diesel Generator A.O. Report and Proposed A.O. Writeup on Indian Point 2 (12 pages)

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POLICY ISSUE
(Information)

January 25, 1984 SECY-84-34

FOR: The Commissioners

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: EMERGENCY DIESEL GENERATORS MANUFACTURED BY TRANSAMERICA DELAVAL, INC.

PURPOSE: To inform the Commission of problems relating to Emergency Diesel Generators manufactured by Transamerica Delaval, Inc. (TDI). We previously provided the Commission with information about TDI diesel generators via Board Notifications 83-160 and 83-160a, and via a memo to Commissioner Gilinsky from Mr. Dircks dated December 14, 1983.

DISCUSSION: On August 12, 1983, the main crankshaft on one of the three emergency diesel generators (EDGs) at the Shoreham Nuclear Power Station broke into two pieces during a load test. The EDGs at Shoreham were manufactured by Transamerica Delaval, Inc. (TDI), which has also provided 54 other EDGs for 14 other nuclear power plant sites in the U. S. A list of those plants is included as the enclosure to this paper. During the course of the evaluation of the failure and the repairs of the Shoreham EDGs, information related to the operating history of TDI engines and the QA program of the manufacturer has been identified which calls into question the reliability of all TDI diesels.

Contact:
R. Caruso, NRR/DL
49-28392

After the Shoreham crankshaft failures, the staff reviewed the operating status of plants where TDI engines had been installed. Grand Gulf, San Onofre 1 (SONGS 1), and Rancho Seco are the only licensed reactors with TDI diesel generators, but SONGS 1 is currently shutdown for seismic modifications, Grand Gulf has only a five percent license, and the engines at Rancho Seco are additional units whose installation is not yet complete. IE Information Notice 83-58 was issued to inform other TDI owners of the failure.

Members of the staff met with representatives of the Long Island Lighting Company (LILCo) several times to discuss the failures to date, the results of LILCo's investigation, and the actions to be taken to recover from the failures. The staff has also developed several lists of questions that it feels need to be addressed as part of the TDI engine evaluations. One list, which has been sent to all TDI diesel owners, requests specific information about each engine. Another was sent to TDI on December 1, 1983, requesting information about the design development history of various parts of TDI machines. Delaval responded on December 16, 1983.

By letter dated December 23, 1983, the staff was informed that a TDI diesel engine owners group has been formed to address these issues.

The staff will be meeting on January 26, 1984 with senior utility executives representing each of the applicants listed in the enclosure. The staff at that time intends to inform the utility management of its major concerns regarding the quality assurance of the TDI design and manufacturing process, and to emphasize the significance of the widespread operating problems to date with TDI engines.

CONCLUSION: For use at the meeting, the staff has prepared a list of operational problems experienced by many TDI engines in both nuclear and non-nuclear service, a summary of the results of the last two QA inspections of TDI, and a summary of the history of TDI vendor inspections since 1979. Taken together with the major crankshaft failure at Shoreham, these operational and QA problems have significantly reduced the staff's level of confidence in the reliability of all TDI diesel generators.

The staff believes that before additional licensing action is taken to authorize the operation of a nuclear power plant with TDI engines, these issues, relating to quality assurance, operating experience, and the ability of the machines to reliably perform their intended function, must be addressed. The staff will report to the Commission on its continued review of this issue.

William J. Dircks
Executive Director for Operations

Enclosure:
As stated

Enclosure

NUCLEAR PLANTS WITH TRANSAMERICA DELAVAL DIESEL GENERATORS

Transamerica Delaval has supplied the DSR and DSRV engines to the following sites:

	Utility	Site	Serial No.	Model
	Long Island Lighting	Shoreham	74010/12	DSR 48
	Middle South Energy	Grand Gulf	74033/36	DSRV 16
	Gulf States Utilities	River Bend	74039/40	DSR 48
	Carolina Power & Light	Shearon Harris	74046/49	DSRV 16
	Duke Power Company	Catawba	75017/20	DSRV 16
	Southern California Edison	San Onofre	75041/42	DSRV 20
	Cleveland Electric Illum.	Perry	75051/54	DSRV 16
	TVA	Bellefonte	75080/83	DSRV 16
	Washington Public Power	WPPSS 1	75084/85	DSRV 16
(*)	Washington Public Power	WPPSS 4	76031/32	DSRV 16
	Texas Utilities Services	Comanche Peak	76001/04	DSRV 16
	Georgia Power	Vogtle	76021/24	DSRV 16
	Consumers Power	Midland	77001/04	DSRV 12
(*)	TVA	Hartsville/Phipps Bend	77024/35	DSRV 16
	SMUD	Rancho Seco	81015/16	DSR 48

(*) project delayed or cancelled

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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cc: CH
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JAN 25 1984

MEMORANDUM FOR: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: ABNORMAL OCCURRENCE REPORT TO CONGRESS
FOR FOURTH QUARTER CY 1983

As requested in your December 30, 1983 memorandum, we have reviewed the items proposed for the Fourth Quarter CY 1983 Abnormal Occurrence Report to Congress. We note that AEOD is not proposing any Abnormal Occurrence at nuclear power plants. NRR would like to propose that the continuing problems with emergency diesel generators manufactured by Transamerica Delaval, Inc. be considered as an Abnormal Occurrence due to major deficiencies in design. In addition, we are in the process of evaluating the inoperable containment spray system event at Indian Point Unit 2. We will inform you later of our findings.

Additional information on the above items along with the input requested in your memorandum is enclosed.

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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ENCLOSURE

PROPOSED ABNORMAL OCCURRENCE

1. Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc. (TDI)

On August 12, 1983, the main crankshaft on one of the three emergency diesel generators at the Shoreham Nuclear Power Station broke into two pieces during a load test. Investigation of this event, along with other operating experience related to diesel generators manufactured by TDI, questions the overall operability of TDI diesels.

A Commission Paper is currently being prepared to address this concern. We will forward a copy of this paper for your use in preparing an Abnormal Occurrence (AO) write-up as soon as it becomes available. We expect that this will occur around January 25, 1984.

2. Inoperable Containment Spray System at Indian Point Unit 2

On November 29, 1983, the containment spray isolation valve on both containment spray trains were found to be closed. A similar event at Farley Unit 2 led to an Abnormal Occurrence finding in the Fourth Quarter CY 1982 Report.

NRR is currently investigating the potential consequences of a limiting LOCA at Indian Point Unit 2 without operation of the containment spray system. If, as in the case of Farley, we find that design conditions could be exceeded, we will propose this event as an Abnormal Occurrence. We will report the results of our findings at a later time.

POSSIBLE OTHER EVENTS OF INTEREST (APPENDIX C TO THE AO REPORT)

1. Fuel Failures (Millstone Unit 2)

This item was originally proposed as an Appendix C item in the Third Quarter CY 1983 AO Report. However, we noted that the licensee's special report was due in mid-November and we recommended that the subject be deferred to the Fourth Quarter.

Attached for your use for an Appendix C write-up are the licensee's November 4, 1983 letter and the staff's Safety Evaluation Report that closed the subject (Attachments 1 and 2).

UPDATING MATERIAL

A. Steam Generator Problems (AO 76-11)

The last update to this item was reported in the Second Quarter CY 1982 AO Report. The write-up was primarily taken from NUREG-0886, "Steam Generator Tube Experience."

AEOD has requested that NRR provide an update to this subject. Efforts are currently underway to publish a supplement to the NUREG. However, the supplement is currently in its first draft and the final package may not be ready for several months. When the supplement is complete we will forward a copy to your staff for use in preparing an update to the Abnormal Occurrence Report.

B. Seismic Design Errors at Diablo Canyon Nuclear Power Plant (AO 81-8)

The last update to this item was reported in the Third Quarter CY 1982 AO Report. AEOD has requested that NRR provide an update to this item.

We agree that significant new information exists to justify an update to the event. Due to scheduling difficulties, we do not foresee completing this update before the end of January 1984. We will forward a copy of this update upon its completion.

C. Large Diameter Pipe Cracking in BWRs (AO 83-5)

As requested by your staff, we have prepared the following update:

The initial report on this topic appeared in NUREG-0090, Vol. 6, No. 3 and described the status as of late October 1983. Previously, BWR pipe cracks generally occurring only in small diameter piping were reported in AO 75-7.

Since the initial report, the four licensees with the following five operating BWRs, Browns Ferry Unit 2, Brunswick Unit 2, Dresden Unit 3, Pilgrim Unit 1 and Quad Cities Unit 2 (plants which had not previously initiated pipe inspections), have now either completed the inspections or, in the case of the Pilgrim plant, have embarked on a pipe replacement program. The Pilgrim licensee, Boston Edison Company, based the replacement decision on the results of a December 15, 1983 inspection program. These five plants were the recipients of the August 26, 1983 BWR Pipe Inspection Orders. The following table presents the results of the inspections at the five plants under the August 26, 1983 Order:

Plant	Welds in Program	No. of Welds Inspected	No. of Indications Detected	No. of Welds Repaired	Restart Date (Approx.)
Br. Ferry 3	191	191	1	1	May 2, 1984
Brunswick 2	131	131	24	9	Dec. 22, 1983
Dresden 3	337	151	28	15	Jan. 20, 1984
Pilgrim		REPLACING PIPE			NOT AVAILABLE
Quad Cities 2	277	246	16	5	Jan. 15, 1984

Georgia Power Company, the licensee for Hatch Unit 2, plans to shut the unit down in January 1984 to initiate a pipe replacement program. The decision to replace pipe was based on the results of the IE Bulletin 83-02 inspection program.

ADDITIONAL INFORMATION REQUESTED FROM NRR

1. License Suspensions

There were no license suspensions during the Fourth Quarter CY 1983.

2. Orders

The following Orders were issued to operating plants during the Fourth Quarter CY 1983:

- Turkey Point Units 3 and 4 (October 14, 1983). Order to change the Technical Specifications to permit continued plant operation in the hot shutdown mode with an RHR pump inoperable.

- Cook Units 1 and 2 (December 16, 1983). Order confirming licensee's commitments on post-TMI issues (II.D.1.2, II.F.1.1, and II.F.1.2).

- Hatch Units 1 and 2 (October 14, 1983). Modification of Order confirming licensee's commitments on post-TMI related issues.

- Peach Bottom Unit 2 (November 30, 1983). Order confirming commitments on pipe crack related issues.

- Rancho Seco (November 10, 1983). Revision of Order confirming licensee's commitments on post-TMI related issues.

- Brunswick Unit 2 (December 12, 1983). Order confirming licensee's commitment on IGSCC inspection.

- Browns Ferry Unit 1 (December 12, 1983). Modification of March 25, 1983 Order confirming licensee's commitments on post-TMI related issues.

- Browns Ferry Unit 1 (December 19, 1983). Order related to pipe crack related issues.

- Vermont Yankee (December 12, 1983). Modification of March 14, 1983 Order confirming licensee's commitments on post-TMI related issues.

3. Identification of Generic Safety Concerns

Resources were allocated to the following generic issues during the Fourth Quarter Cy 1983:

Generic Issue #	Subject
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors
61	SRV Discharge Line Break Inside the Wetwell Airspace of BWR Mark I and Mark II Containments
66	Steam Generator Requirements
69	Make-up Nozzle Cracking in B&W Plants
75	Generic Implications of ATWS Events at the Salem Nuclear Plant
82	Beyond Design Basis Accidents in Spent Fuel Pools

ATTACHMENT 1

NORTHEAST UTILITIES
THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

November 4, 1983
Docket No. 50-336
A03429

Director of Nuclear Reactor Regulation
Attn: Mr. James R. Miller
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) R. A. Clark letter to W. G. Counsil, dated July 29, 1983.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Failed Fuel Assemblies

A meeting was held in the Bethesda Offices of the NRC Staff on October 12, 1983, to discuss the fuel pin failures and fuel assembly failures observed in the Millstone Unit No. 2 fuel. At that meeting, Northeast Nuclear Energy Company (NNECO) also identified the schedule for submitting revised safety analyses to support Cycle 6 operation with a revised loading pattern.

As a followup to the meeting, NNECO hereby documents in Attachment 1 the materials presented at and discussed during the October 12, 1983 meeting. The following is a summary of the October 12, 1983 presentations.

Leaking Fuel Pins

At the end of Cycle 5 operation, the primary system activity was approximately two percent of the Technical Specification limit, this is indicative of ten to thirty fuel pin failures. As a result of this activity, a fuel pin failure investigation program was established. The program included fuel assembly sipping, visual examinations, ultrasonic examinations of fuel pins , a review of the plant operating history and a review of the manufacturers records. Westinghouse Electric Corporation, the fuel vendor, Northeast Utilities Service Company and Northeast Nuclear Energy Company are the major parties involved in the ongoing investigation and various tasks are being performed as joint efforts. Typical of the joint efforts are the visual inspections, fuel assembly sipping and fuel rod ultrasonic testing. In addition to the efforts described above, Westinghouse Electric Corporation has assembled a multidiscipline team to address the fuel pin problems and potential solutions. A senior fuel performance person from Northeast Utilities has been assigned to the Westinghouse Electric Corporation team as a participant.

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Fuel sipping, utilizing a wet process, was conducted on the entire core and revealed twenty-six fuel assemblies with failed clad. The twenty-six assemblies which are classified as leakers consist of twenty-one Westinghouse supplied assemblies and five Combustion Engineering assemblies. The Combustion Engineering assemblies are scheduled for discharge while the Westinghouse assemblies were scheduled for reinsertion.

Visual examinations were conducted utilizing underwater television and a periscope. In conjunction with the visual examinations many fuel rods were lifted to allow examination of the fuel rod grid area. The results of the visual examinations are shown in Table 1. Many of the observations occurred in the area of the first grid or at the upper end of the fuel rods.

Ultrasonic examinations were conducted on all fuel rods in each failed fuel assembly as well as some non leaking assemblies. The ultrasonic inspection revealed 32 failed pins in the twenty six leaking fuel assemblies. The identification of the failed pins allowed additional evaluation of manufacturing records and any geometry related anomalies. The initial search of the manufacturing records revealed no significant inter-relationship. The examination did reveal some paired pin failures which indicates the potential of debris related failures..

A number of rods were scraped and material was collected for analysis which is still underway. The rod scraping did reveal that the reflective patches on the end plug welds were adherent to the fuel rod surface and had no apparent depth.

The basic approach taken in the failure evaluation was to evaluate all possible failure mechanisms and to then classify the various mechanisms as to the probability of being a contributor. Based on this evaluation, the potential mechanisms which received the most attention are debris-induced fretting, welding defects, and grid spring/rod fretting. The mechanisms which could not be eliminated, but appear less probable based on current data, are fuel hydrogen or microstructure effects, cladding defects and pellet/clad interaction. Additional data is needed to more conclusively evaluate these mechanisms. Mechanisms considered unlikely are corrosion, crud, fatigue, rod bow, and densification/collapse.

One confirmed case of grid/rod fretting was observed in a Region 7 fuel assembly. Additional rods were lifted (~100) in order to provide some assessment of the extent of this mechanism. No additional cases of this mechanism were observed. The probable cause of the one observed fretting failure is a damaged cell, most probably caused by either manufacturing or handling.

The evidence suggesting debris induced wear are the many debris observations in the core, the two incidents of paired pin failures, and some evidence of a wear scar on one rod.

The concern for welding related defects stems from the observation of the patches at the top end plugs. The best assessment at this time is that these areas have no obvious depth and are "cosmetic" in nature. However, the observations are sufficiently frequent that further evaluations are needed.

In summary, the fuel pin failures identified apparently result from multiple sources, none of which are indicative of a situation that may lead to continued serious degradation of the fuel clad. The question of primary system debris is being addressed by an extensive primary system clean up campaign which includes the reactor vessel, internals and fuel. The grid spring/fuel rod fretting is not considered to be flow induced, but is handling or manufacturing related and is expected to be very limited. Other failure mechanisms are still under investigation but no potential causes have been identified that would lead to wide spread degradation of the clad.

All failed fuel assemblies, except one assembly that was classified as a "probable leaker", were eliminated from the revised Cycle 6 loading pattern.

Broken Holddown Springs

The results of the holddown spring inspections indicated that there are eight (8) batch G assemblies and seven (7) batch F assemblies each with one broken holddown spring. Fifty (50) batch E assemblies were inspected which included all peripheral Cycle [illegible] assemblies and two (2) additional assemblies. There were no broken springs found in batch E. Additionally, twenty (20) batch C, Cycle 1, peripheral assemblies were inspected with no failures found.

All broken springs were located on fuel assemblies adjacent to the core periphery in either Cycle 4 or Cycle 5 and exhibit high cycle fatigue with the exception of the broken spring in assembly F73. Upper nozzle post wear behind the coil springs has been observed in failed and unfailed spring locations in batch G and F assemblies. This is observed in interior locations as well as peripheral core locations. There are no indications of loss of assembly holddown through inspections of the lower core support plate, selected fuel assembly grid locations, and the upper nozzle posts of the affected assemblies.

The probable cause of the holddown spring failure is system flow induced vibration, near the periphery, leading to fatigue failure. The local flow conditions at the peripheral core locations in combination with the steady state stresses in the spring apparently lead to the failure of a small fraction of these springs. The impact of these breaks was quite small. All but one spring broke at the beginning of the first active coil (either top or bottom), as would be expected analytically. The spring in assembly F59 is broken in the middle as opposed to the beginning of the active coil. It is postulated that this one spring may have had a flaw or material defect which results in local stress concentration at the break location.

Although broken, the springs remain functional but with smaller free lengths. For the observed breaks, the design holddown requirements are met even with multiple broken springs within one assembly since the loss in holddown capability of a broken spring is small. Additional breaks in a previously broken spring are unlikely due to the reduced stress in the spring following the break. This has been confirmed by the visual examinations where no spring was observed to have multiple breaks.

The maximum observed two cycle nozzle post wear was less than one-quarter of the post thickness (~270 mils.). The main structural member of the assembly is the guide tube which extends up through the guide post. Therefore, significant wear in excess of that already observed could be sustained without affecting the structural integrity of the assembly and continued operation with these assemblies is acceptable.

<u>Fuel Assembly Structural Damage</u> (F73 and F37)

Assembly F73 was discovered to have one flower petal bent relative to the remainder of the flower. This petal is apparently holding the flower depressed below its normal rest elevation. There is an indentation mark on the top of the bent petal believed to be caused by the interaction between the flower and fuel alignment plate. The spring 180° opposite to the bent petal is broken. This spring is the only broken spring which was never located at the core periphery. It is postulated that since the flower is depressed below its normal rest elevation and also below its normal hot full power elevation, the steady state stress in the spring at operating conditions was larger than the other springs in the core. The springs in assembly F73 were not loaded uniformly since the flower is no longer parallel to the adaptor plate. Although the flow conditions experienced by this spring were not the same as the peripheral springs, it is expected that the larger, mean stress in combination with the local flow led to a low cycle fatigue failure.

Assembly F37 was discovered to have local buckling of one guide tube at the top of the upper swage joint where the grid sleeve is attached to the guide tube. The adaptor plate at that same corner of the assembly is bent and there are score marks on the slot surface of the nozzle post at the same assembly location.

The probable cause of the damage to both F73 and F37 is attributed to restricted flower movement relative to one guide post and the interaction of the fuel alignment plate subsequently damaged the two fuel assemblies.

The fuel alignment plate interfaces with the flower in two different ways depending upon whether the assembly is in a Control Element Assembly (CEA) or non-CEA location. In a CEA location (F73), only the outer petals of the flowers are contacted by the core plate so that restricted motion will result in bending of the flower. It has also been analytically verified that the flower petal will bend under excessive force before other damage occurs.

In a non-CEA location (F37), a large portion of the flower and the ligaments between the guide post holes are contacted by the fuel alignment plate so that restricted motion will result in local buckling of the guide tube.

CEA motion is unimpaired in both damage scenarios. No anomalous CEA behavior was noted during Cycle 4 or Cycle 5 operation. Free CEA motion has been verified in both assemblies in the spent fuel pool subsequent to operation in the core during Cycle 5. Analytical verification has also been provided to show that for the type of guide tube damage observed, the guide tube will buckle radially outward so as not to restrict CEA motion. The inner diameter of the nozzle post is also unaffected following the observed flower damage (F73) due to its large thickness. The structural integrity requirements, such as strength and loading capability, of the damaged fuel assemblies meet normal as well as design accident conditions. Fuel subcomponent testing has been performed in the past in order to verify these analytical results. Assemblies F73 and F37 functioned acceptably during Cycles 4 and 5.

Planned Actions

All new fuel assemblies will be reworked with new flowers and springs. The new springs will operate with significantly less steady state shear stress thus increasing the margin to fatigue failure. The new springs have a smaller post to spring gap thus reducing the amplitude of lateral deflection. Additionally, the higher natural frequency of the new springs will help reduce excitation phenomenon such as base motion or vortex shedding. The new flower/pin design will preclude pin to slot contact under the worst design tolerance conditions. Additional clearance is also provided between the cylindrical surfaces of the post and flower and the flower hole diameters are chrome plated. The new spring characteristics will be verified with flow and mechanical testing.

Assemblies F73 and F37 will be removed from the revised Cycle 6 loading pattern. A flower compression/load test will be performed on all irradiated fuel in the spent fuel pool prior to insertion in Cycle 6 to verify freedom of flower movement as well as the load required. Since it has been analytically verified that any assembly with internal damage to a guide tube would also exhibit damage at the upper swage joint or flower (e.g. F37 and F73), no further visual inspections of the fuel is necessary.

During the meeting, the Staff requested additional information which we committed to provide.

- Attachment 2 represents NNECO's response to the Reference (1) Staff inquiry regarding the failed fuel at Millstone Unit No. 2.
- As requested, a tabulation of all licensee event reports (LER's) generated as a result of exceeding the technical specification limit on reactor coolant iodine concentration is provided in Table 2.
- NNECO intends to submit the plans for post Cycle 6 fuel surveillance 90 days prior to the Cycle 7 refueling shutdown.

We trust you find this information satisfactory and responsive to your requests.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil
Senior Vice President

Table 1

TABULATION OF VISUAL OBSERVATIONS

1. Hydrided Fuel Rods: Five rods with hydride patches were observed, four in Batch F and one in Batch G. Several of these rods had other anomalies.

2. Missing end plug: One rod with a missing plug was observed.

3. Fission Product Plumes/Grid-Rod Fretting: Three rods had plumes coming from the lower grid area. All of these rods were lifted. One showed a clear case of grid/rod fretting.

4. "White Spots" Behind Grids: "White spots" behind the grid springs were observed quite frequently. Several of these rods have been lifted and show blister or scar like observations.

5. Wear scar: A wear scar was observed below the first grid on one rod. The rod was not obviously failed at that spot in the portion visible. The rod showed hydride patches further up the road.

6. Reflective patches at top end plug: A high frequency of reflective patches at the top end plugs was observed. The reflective areas appeared patchy in nature and have no obvious depth.

7. Debris Sightings: Debris was observed in both failed and non-failed assemblies, and the majority of the observed debris was non-metallic.

8. Clustered Rod Failures: One group of three failed rods and one possible case of paired failures were observed.

TABLE 2

MILLSTONE UNIT NO. 2
LICENSEE EVENT REPORTS
ASSOCIATED WITH LEAKING FUEL

NUMBER	DATE
50-336/82-19	June 9, 1982
50-336/82-38	September 18, 1982
50-336/82-50	December 31, 1982
50-336/83-03	February 19, 1983
50-336/83-19	May 28, 1983

ATTACHMENT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 19 1983

MEMORANDUM FOR: [redacted], Assistant Director
for Operating Reactors, DL

FROM: L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

SUBJECT: SAFETY EVALUATION FOR MILLSTONE UNIT 2, CYCLE 6 RELOAD
(TAC 49798)

Plant Name:	Millstone Nuclear Power Station Unit 2
Docket Number:	50-336
Licensing Stage:	Operating Reactors
Responsible Branch:	Operating Reactors Branch #3
Project Manager:	K. Heitner
Description of Review:	Reload SER
Review Status:	Complete

Attached is the Core Performance Branch safety evaluation input for the Cycle 6 reload of Millstone Unit 2. We conclude that reasonable assurance has been provided that Cycle 6 operation of Millstone 2 will not pose a threat to the health and safety of the public and that the proposed operation is, therefore, acceptable.

L S Rubenstein
L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

Attachment:
As stated

cc: R. Mattson
D. Eisenhut
J. Miller
K. Heitner
R. Capra
O. Parr

Contact: M. Dunenfeld, CPB:DSI
X-28097

A. Gill, CPB:DSI
X-29071

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SAFETY EVALUATION REPORT

MILLSTONE UNIT 2, CYCLE 6 RELOAD (TACS 49798)

1. INTRODUCTION

In Reference 1, Northeast Nuclear Energy Company (NNECO) submitted a license amendment request and the Reload Safety Analyses (RSA) in support of the Millstone Unit No. 2, Cycle 6 reload. As indicated in the submittal, the bases on which the Cycle 6 reload was analyzed were documented in a "Basic Safety Report" (BSR) (Ref. 2). The BSR, as supplemented by Reference 3, serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone 2 (a Combustion Engineering plant). Reference 4 documents the NRC staff's review and acceptance of the BSR.

By Reference 5, NNECO informed the Staff that due to the elevated levels of radioactive iodine and other fission products identified during Cycle 5 operation, NNECO anticipated the discovery of a number of fuel assemblies with leaking fuel rods during the 1983 refueling outage.

Since that time, NNECO performed fuel sipping identifying 26 fuel assemblies with failed fuel rods. In addition, visual examinations revealed 15 fuel assemblies to have broken holddown springs. Further, structural damage was observed in two assemblies, one of which also had a broken holddown spring. This damage was reported to the Staff in License Event Reports 50-336/83-25, 83-25/01-T, 83-26, and 83-26/01-T. Reference 5 provided a detailed discussion of the fuel degradation.

As discussed in Reference 6, NNECO is replacing all leaking fuel assemblies with a combination of new and previously discharged fuel assemblies. These changes have necessitated a revised loading pattern for Cycle 6 operation. In addition, assemblies F37 and F73, which sustained some structural damage, are being replaced.

By Reference 7, NNECO reported damage to the thermal shield support system at Millstone Unit No. 2. The extent of this damage resulted in the need for removal of the thermal shield from the core barrel. Reference 8 provides details of NNECO's thermal shield damage recovery program.

In order to assess the impact of a new loading pattern and the removal of the thermal shield, NNECO has had its fuel vendor reevaluate the Reference 1 Reload Safety Analyses in support of Millstone Unit No. 2 Cycle 6 operation. The results of this review were provided as a supplement to the Reload Safety Analyses (Reference 6).

1.1 General Description

The Millstone 2 reactor core is comprised of 217 fuel assemblies. Each fuel assembly has a skeletal structure consisting of five (5) Zircaloy guide thimble tubes, nine (9) Inconel grids, a stainless steel bottom nozzle, and a stainless steel top nozzle. One hundred seventy-six fuel rods are arranged in the grids to form a 14x14 array. The fuel rods consist of slightly enriched uranium dioxide ceramic pellets contained in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel.

Nominal core design parameters utilized for Cycle 6 are as follows:

Core Power (Mwt)	2,700
System Pressure (psia)	2,250
Reactor Coolant Flow (GPM)	350,000
Core Inlet Temperature (°F)	549
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stack height of 136.4 inches)	6.065

The core loading pattern for Cycle 6 consists of twenty-four (24) interior feed assemblies containing 2.7 w/o U-235 and forty-eight (48) peripheral feed assemblies containing 3.2 w/o U-235. These are replacing seventy-two (72) Combustion Engineering (CE) batch E assemblies. Due to fuel defects in Cycle 5 and subsequent symmetry consideration, sixteen (16) interior feed assemblies containing 2.70 w/o U-235, twenty (20) CE assemblies from Batch A and one (1) CE assembly from Batch B (these CE assemblies were discharged at the end of Cycle 1) are needed as well.

2. FUEL SYSTEM DESIGN

The fuel system design for Millstone Unit 2, Cycle 6 is the same as that approved (Ref. 4) for Cycles 4 and 5. That is, approval of the BSR constituted approval of the use of a mixed core of Combustion Engineering and Westinghouse fabricated fuel assemblies. The replacement of CE fuel with Westinghouse fuel at each reloading would eventually lead to a core with all Westinghouse fuel.

The failed fuel assemblies at Millstone necessitated a revision to the reload plan such that a mixed core, as described in Section 1.1 results. The reload redesign is a result of the following:

1. Fuel rod failures in 26 assemblies
2. Removal of two damaged fuel assemblies
3. Removal of the thermal shield
4. Failure of holddown springs in 15 fuel assemblies.

As described in Reference 9, the reload redesign utilizes a combination of new and previously discharged fuel assemblies to replace the leaking and broken fuel assemblies. Since this redesign uses previously approved fuel assembly types, and since the redesign and the reinserted CE assemblies will not receive greater than design exposure the redesign is acceptable from the fuel system point of view.

The NRC was informed of the broken holddown springs identified on 15 fuel assemblies by Reference 10. A summary of information discussed at a meeting on October 12, 1983 on the broken holddown springs was presented in Reference 6. At this meeting, NNECO documented plans to evaluate the replacement of the broken holddown springs. A repair procedure and tooling was developed to effect the replacement of the holddown springs on irradiated fuel assemblies. This procedure was utilized successfully on one fuel assembly. However, NNECO decided that the irradiated fuel repair procedure involved a high risk with the potential for damaging fuel assemblies, particularly fuel pins, during the repair.

NNECO therefore reached the conclusion and provided supporting analysis (Ref. 11) that operation of Cycle 6 with 9 fuel assemblies, each with a single broken holddown spring, is acceptable and prudent. The analysis provided by NNECO characterizes the breaks to the holddown springs, provides justification that the breaks were caused by excessive vibratory motion during reactor operation, discusses fretting wear, loose parts, control rod jamming and the probability of multiple fractures, and concludes that operation of Cycle 5 with the 9 assemblies having broken holddown springs would be acceptable. This is primarily because the number of active turns of the springs is only slightly decreased by the types of breaks observed. Future new fuel will have newly designed springs.

We have reviewed the material provided by NNECO and agree with the conclusion that operation of Cycle 6 with 9 assemblies containing broken holddown springs will not pose a significant reduction in safety of the power plant.

3. NUCLEAR DESIGN

The nuclear design procedures and models used for the analysis of the Millstone Unit 2 Cycle 6 reload core (Reference 1) are the same as those used for Cycle 5. These are documented in the Millstone Unit 2 Basic Safety Report (BSR) (Reference 2) and have been approved (Reference 4) for the analysis of the Millstone Unit 2 core using Westinghouse reload fuel beginning with Cycle 4.

The licensee provided a tabular summary (Table 2, Reference 1) of the changes in the Cycle 6 kinetics characteristics compared with the current limits based on the most limiting BSR safety analysis and the Cycle 4 and 5 analyses. All of the Cycle 6 values fell within the current limits. The kinetics parameters were therefore acceptable for use in the Cycle 6 accident analysis, because they are calculated with approved methods, and they are within the bounds of values previously approved.

The reanalysis of the reload performed as a result of the fuel failures (Reference 9) and removal of the thermal shield was performed with the same approved techniques discussed above. In Reference 9, Table 2 the kinetics parameters for the Cycle 6 reload redesign are given. These are all within the current limits with a small exception in the least negative and above 30% power doppler temperature coefficients and the maximum delayed neutron fraction. The licensee examined the effects of these changes on accident analyses in Reference 9, pages 7 and 8, with the conclusion that the potential effects were small, and no reanalyses were necessary. We reviewed these evaluations and agree that the small changes in these parameters do not lead to a need for reanalyses of any accidents, and that the revised fuel loading and removal of the thermal shield is acceptable with respect to nuclear design.

The control rod worths and shutdown requirements for the Cycle 6 redesign and the initial Cycle 6 design are presented in Table 3 of Reference 9 and compared with previous Cycle 5 values. At EOC 6, the reactivity worth with all control rods inserted assuming the highest worth rod is stuck out of the core is 6.00% assuming a 10% reduction to allow for uncertainty. The reactivity worth required for shutdown, including the contribution required to control the steamline break event at EOC 6 is 5.92% . Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break at the worst time in core life allowing for the most reactive control rod stuck in the fully withdrawn position and also allowing for calculational uncertainties. We have reviewed the calculated control rod worths and the uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR and in previous Westinghouse reports. On the basis of our review, we have concluded that the NNECo's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod is stuck in the fully withdrawn position.

The total trip reactivity as a function of position calculated for Cycle 6 was more limiting than that calculated for Cycle 5. The Cycle 6 curve was therefore used in all accident reanalysis.

4. THERMAL-HYDRAULIC DESIGN

Millstone 2 Cycle 6 utilized the Basic Safety Report (Ref. 2) which was approved by the staff in Reference 4. The Basic Safety Report was also used as the basis for Cycles 4 and 5 operation.

As discussed in the BSR, the Westinghouse fuel assemblies have been designed and shown through testing to be hydraulically compatible with all resident Millstone 2 fuel assemblies. A detailed discussion is given in the staff SER of Cycle 4 dated October 6, 1980 (Ref. 12).

The DNB analysis for Cycle 6 was performed for a minimum reactor coolant flow rate of 350,000 gpm and a radial peaking factor, F_r, of 1.565. A reduction in flow from 370,000 gpm to 362,600 gpm and a conservative reduction in F_r from 1.63 to 1.597 was previously implemented during Cycle 5 operation. As indicated by the power and flow sensitivities reported in the Cycle 4 Reload Safety Evaluation Report (Ref. 13) a flow reduction can be offset by a power (or F_r) reduction in a 2:1 ratio to maintain a constant DNBR. Thus the reduction in flow has been more than offset by the reduction in radial peaking factor and this has been confirmed by the licensee in their Cycle 6 analysis. The Cycle 6 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of various measured plant power parameters in terms of power. This partial credit was applied in previous cycles and its approval is discussed in more detail in the Cycle 4 Reload Safety Evaluation Report (Ref. 13); therefore, we find operation of Cycle 6 acceptable.

5. ACCIDENT ANALYSIS

The transients and accidents and LOCA analyses will be provided by the Reactor Systems Branch separately except for the CEA withdrawal which is discussed below.

5.1 CEA Withdrawal at Power

The CEA withdrawal at power accident was reanalyzed for Cycle 6 to assess the impact of increased steam generator tube plugging and the corresponding reduction in flow. The results of this analysis show that the thermal margin low pressure trip maintains the minimum DNBR above 1.30 over the full range of reactivity insertion rates, which is acceptable.

6. TECHNICAL SPECIFICATIONS

Technical Specification changes proposed by the licensee in Reference 1 are acceptable as follows. No additional Technical Specification changes were required as a result of the reload reanalysis.

A. Reduced Reactor Coolant Flow Rate - This proposed change affects pp. 2-2, 2-4, and 3/4 2-14 of the Technical Specifications. It involves lowering the required primary coolant flow rate from 362,500 gpm to 350,000 gpm. This new lower flow is established to correspond to a plugging level of 2500 steam generator tubes, and was used in the Cycle 6 analysis. We find it acceptable since it was offset by the reduction in F_r.

B. CEA Drop Time - This proposed change to p. 3/4 1-26 of the Technical Specifications involves a revision of the CEA drop time. At the beginning of Cycle 3, four small flow hole test assemblies were put into the core under CEA locations in an effort to mitigate the guide tube wear problem. At that time, the CEA drop time was changed from 2.75 seconds to 3.1 seconds due to a larger dashpot effect realized with the reduced flow holes. This design is no longer being used as the "guide tube wear" fix at Millstone Unit 2 and the four test assemblies will be removed from the core during this 1983 refueling. The licensee, therefore, proposed changing the CEA drop time back to the original value.

C. New Axial Shape Index Tent - The change to p. 3/42-4 involves a new axial shape index (ASI) monitoring tent for figure 3.2-2 of the Technical Specifications. This tent is used to verify the kw/ft limit of 15.6 which is input to the LOCA analyses. Operation within the tent ensures that the maximum local power is less than 15.6 kw/ft. and thus satisfies the Technical Specification surveillance requirement. Under normal conditions the kw/ft surveillance limit is verified with the incore monitoring system and the only time the ASI tent is used is if the incore system is inoperable.

D. Revised total planar peaking factor, F_{xy}, curve - This change affects pp. 3/4 2-6 and 3/4 2-8 of the Technical Specifications and involves restoring the planar radial peaking factor, F_{xy}, monitoring limits back to the original Beginning of Cycle (BOC) 5 values. The Cycle 6 licensing analyses support this proposed revision.

E. Revised total radial peaking factor (F_r) curve - This proposed change affects pp. 3/4 2-8 and 3/4 2-9 of the Technical Specifications. In comparing the BOC 5 values to BOC 6 values, the required primary flow is being reduced by 5.4% (370,000 gpm to 350,000 gpm). Although the current licensed primary coolant flow rate is 362,600 gpm, BOC 5 values are being used since these values correspond with those of the last transient analysis. The Cycle 4 Reload Safety Analyses have shown that the DNB analysis penalty which results from a reduction of 2% in primary flow can be offset with an approximate 1% reduction in F_r. Therefore, the 4% reduction in allowable F_r more than offsets the penalty associated with a 5.4% reduction in primary flow. The Cycle 6 licensing analyses support this proposed revision.

F. Auxiliary Feedwater Pumps - These proposed changes make Millstone Unit 2 Technical Specifications, specifically p. 3/4 7-4, consistent with NUREG-212, Revision 2 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors. In addition, the proposed revision modifies the Technical Specifications to reflect the actual plant conditions applicable to Mode 4 under which there is insufficient steam to allow the steam turbine driven auxiliary feedwater pump to meet the required discharge pressure.

These changes are all acceptable because they are consistent with the Cycle 6 licensing analysis, or, in the case of the latter item, make the Millstone Unit 2 Technical Specifications consistent with the accepted specifications of NUREG-212.

7. CONCLUSION

We have reviewed Millstone Unit 2 Cycle 6 reload and the proposed changes to the Technical Specifications and find they pose no significant hazard. The reload uses approved fuel types and will not cause any change in the types or increase in the amount of effluents or any change in the authorized power level of the facility. The transients and accidents, and provisions for reactivity control meet applicable criteria. The amendment therefore does not:

(a) involve a significant increase in the probability or consequences of an accident previously evaluated; or
(b) create the possibility of a new or different kind of accident from any accident previously evaluated; or
(c) involve a significant reduction in a margin of safety.

8.0 REFERENCES

1. W. G. Counsil (NNECO), letter to R. A. Clark (NRC), with Millstone Unit 2 Reload Safety Analysis, April 13, 1983.

2. "Basic Safety Report," Westinghouse proprietary report for Millstone Unit 2, Docket Number 50-336, submitted via letter, W. G. Counsil (NU) to R. Reid (NRC), March 6, 1980.

3. W. G. Counsil (NNECO), letter to R. A. Clark, November 17, 1981.

4. L. S. Rubenstein (NRC), memorandum for T. M. Novak, "SER Input on Millstone Unit 2 BSR," February 16, 1982.

5. W. G. Counsil (NNECO), letter to R. A. Clark (NRC), June 22, 1983.

6. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), November 4, 1983.

7. E. J. Mroczka (NNECO), letter to T. E. Murley (NRC), July 1, 1983.

8. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), September 15, 1983.

9. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), November 17, 1983.

10. E. J. Mroczka (NNECO), letter to T. E. Murley (NRC), August 12, 1983.

11. W. G. Counsil (NNECO), letter to J. R. Miller (NRC), December 1, 1983.

12. R. A. Clark (NRC), letter to W. G. Counsil (NNECO), October 6, 1980.

13. W. G. Counsil (NNECO), letter to R. A. Clark, June 3, 1980.

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HRD

REF. NO.

February 1, 1984

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

Subject: TDI - Emergency Diesel Generators

Dear Mr. Denton:

I was unable to attend the meeting on January 26, 1984, concerning the emergency diesel generators manufactured by Transamerica Delaval, Inc. At the time I was on board one of our "semi-private islands" somewhere on the Mississippi River. In the interval, I have had an opportunity to review selective reports concerning the meeting and plan to obtain, when available, a transcript.

The Falcon Shipping Group owns eleven American flag vessels, five of which utilize TDI engines for the main propulsion. The first of our vessels with Delaval engines, the PRIDE OF TEXAS, was delivered to us in May 1981. A descriptive brochure is enclosed. As your staff is aware, we have been experiencing a number of serious and yet unexplained engine problems. Enclosed is a letter dated January 5 to TDI which reviews one such problem. Since our business depends on these engines performing successfully, we are also interested in and committed to a program that seeks to determine the underlying causes for such problems. An investigation of the design reliability will not be an easy task and may confirm, in the end, our worst fears. Yet there is an equal chance that such an investigation will remove the clouds of doubt that now exist and permit each engine owner to better predict and rely on the engine performance.

Having met or talked with individual members of the nuclear utilities, I believe that they are also committed, on the whole, to a program that removes these doubts. However, I also note my sense of the enormous economic pressures present that will attempt to limit the depth and scope of such an investigation. I personally believe in the commercial utilization of nuclear energy -- utilization that is carefully regulated and monitored. Nuclear energy has an enormous potential for both good and bad. If the benefits

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January 5, 1984

Ref. No.

Clint Matthews, General Manager
Transamerica Delaval, Inc.
Engine & Compressor Division
P.O. Box 2161
Oakland, CA 94621

Subject: Texas Class Vessels - Piston Failures

Dear Clint,

During my visit with you in November of 1983, I promised you a detailed summary of the facts and costs concerning the piston failures experienced to date on the Texas Class Vessels. Hopefully you will have an opportunity to review this prior to our meeting in New York City on January 6,1984. In addition, I would like to take this moment to express to you my personal concerns, fears and conclusions concerning the problem and its potential implications to our business.

The tale I intend to unfold is not a pleasant one nor is it a short one. Many sections reflect my opinions and observations, and it is certainly not intended to be technical or an authoritative statement on the design problems of medium speed diesel engines. In this regard, Falcon has and continues to rely on Transamerica Delaval to stand behind its representations concerning these engines. We are shipowners and not engine manufacturers. Since the occurrence of the casualty on the Star of Texas on October 8, 1983, I have attempted to personally vicit each ship to conclusion is the fact that for at least a one year period, Service Representatives from Delaval were riding and observing at least one or more of the vessels on a continuous basis. During this period of time, nothing was brought to my attention that would indicate unreasonable or negligent operation. This is not to say that occasional problems did not then or do not now exist, but that <u>on the whole</u> the engines are being operated in a professional and responsible manner. Finally, I do not attribute the piston failures to operating or maintenance practices because the piston defect is also present in Enterprise engines that are land based (i.e. no propeller) operated exclusively on diesel oil and operated only occasionally for short periods of time (i.e. Nuclear power plant generators).

One of my objectives in writing this letter to you is an honest attempt to resolve this problem without resorting to time-consuming and costly litigation. As businessmen, there must be some type of compromise which we can reach on this controversy. We are not asking Delaval to admit liability but only to make an honest appraisal of responsibility. Any settlement can be worded to preclude any party from admitting liability. During the course of our conversations, you have noted many times the terms of the "Warranty". As a consequence of this warranty, you have taken the position that Delaval's sole responsibility is to replace the defective part (within the warranty period) but that labor, delay and other costs are outside the terms of your responsibility. On many occasions I have also noted Falcon's

This telex followed a meeting in Oakland where the original piston problem was discussed and I again was assured that the modifications made to the pistons on the Star of Texas in Haifa (August 1982) were adequate and satisfactory. At no time was any other problem with the pistons mentioned or noted.

Yet now based on subsequent information, I discover that before March 15, 1983, Transamerica had actual knowledge that another piston defect existed. If we had been told of this manufacturing problem at an earlier date we could have taken steps to replace the pistons during the drydocking of the Star of Texas in July of 1983. Instead, a piston separated while the vessel was enroute to Alexandria, Egypt in October 1983. The vessel was forced to deviate to Halifax, Nova Scotia where extensive and costly repairs were undertaken. It was only by sheer luck that we avoided cracking the engine block. Changing out the pistons during drydocking would have avoided this casualty and the additional five days of delay suffered at Tampa during the actual changeout.

There may be some merit and some justification for a company to attempt to limit its liability under a warranty provision. However, when a company such as yours has actual knowledge that a manufacturing problem exists and fails to take reasonable and responsible steps to notify its customers, then I believe that such a company should be held fully responsible for the consequences. Transamerica Delaval has the changeout, the vessel was delayed and $46,000 of outside labor was incurred. Under repeated questioning, we were assured by Transamerica Delaval that the problem had been corrected and that the pistons were fit for their intended purpose.

PRIDE OF TEXAS - TUNISIA

Upon arrival in Tunisia in August 1982, work immediately commenced to remove the defective pistons and grind out, by hand, the stress-raiser. The vessel suffered additional costs as a result.

SPIRIT OF TEXAS - JAN. 1983

The piston changeout was extensively discussed prior to delivery of the Spirit of Texas in January of 1983. In order to reduce the costs to Delaval, Falcon agreed that the piston changeout would occur after delivery. Therefore, as the Spirit of Texas was loading its first cargo of bagged flour in Beaumont, Texas, Delaval undertook the piston changeout. Unfortunately, during this period of time, I was in Egypt and had no actual knowledge of the circumstances surrounding the changeout. However, it is now clear that by at least January of 1983, Delaval had actual knowledge that another problem existed with the pistons. Without advising Falcon or Titan, Delaval replaced the existing pistons with another set of pistons which had been stress relieved in accordance with the new manufacturing procedure. This was discovered when Delaval informed us, on November 2, 1983, that the pistons on the Spirit of Texas did not have to be corrective action at Haifa, or an isolated occurrence. Again we note for your attention Attachment No. 12 which is a photograph of the replacement piston. There appear to be significant differences in the boss area design as compared to other pistons. Can you explain the reason for this difference?

EARLIER PISTON CASUALTY

During the course of my investigation, I also discovered that in May 1983, the Star of Texas had suffered an earlier piston failure. In that case the crown had not separated from the skirt. The cracking was along the skirt and repairs were undertaken at sea. The Chief Engineer concluded that it was only an isolated incident.

SUBSEQUENT MEETINGS

During the course of our inquiry into the cause for the October 1983 piston failure on the Star of Texas we were informed by Transamerica Delaval that another piston defect existed. At a meeting in Houston on November 2, 1983, you discussed the background behind the defect and the steps that your company had taken to correct the defect. Apparently, when the piston skirts were originally manufactured they were not subject to an adequate stress relieving process. We were told the original process involved heating the skirt to 1,700 degrees for a period of three hours and then forced air cooling. In the new procedure, the skirt is then reheated for a period of three hours at 1,050 degrees followed by natural cooling. It is this second reheating and cooling that is the critical step in stress relieving the skirts.

the foundry and the new stress relieving procedure was again outlined. During the course of the visit I became aware that the replacement stress relieved pistons were in most cases used pistons. It was my understanding that as Delaval removed defective pistons from various customers that they were shipped back to Oakland for inspection and stress relieving. This replacement program with used pistons from other customers raises some serious questions. First, I am concerned that your present inspection and testing techniques do not discover latent defects in the used pistons. Specifically, Delaval has acknowledged the high risk of cracks originating in non-stress relieved pistons. As a consequence, the fact that these used pistons were in operation may have generated cracks not discovered by your Company. Therefore, stress relieving such pistons will ultimately be of no benefit.

Another problem I have with this replacment program is the lack of documentation provided to us on the origin and utilization of these used pistons. I would appreciate documentation on the origin of each of the pistons delivered to us, and where possible the number of hours of service accumulated on each piston. Finally I would also appreciate a report from you on the procedures taken to inspect these pistons.

STAR OF TEXAS - TAMPA, FLORIDA

The Star of Texas arrived at Tampa, Florida, on November 28, 1983. After clearing customs at approximately 7:00 p.m., skirt is unpredictable. It is highly unlikely that all these cracked pistons could have occurred at the same time. My concern over the origin of the replacement pistons was increased as a result of certain findings. On November 30, 1983, the eddy current examination of replacement piston ID #X38-782G-SR revealed a small crack at the intersection of the leg and boss area around the bolt hole. This crack was in the order of 1/34 of an inch in length. With a later dye check of this area, it was possible to visually see the crack. On that date, I immediately telexed the following to you:

> "I am obviously concerned with this finding. In releasing this piston to us, your quality control people may have concluded that this crack was insignificant. If so, we will rely on their judgement. On the other hand, it may be an originating crack that over time may grow. Your comments and conclusions concerning this piston would be appreciated. Unless I hear to the contrary, we will continue with the piston changeout based on a belief that your company stands behind its work and representation that these stress relieved pistons are adequate for their intended purpose".

In response Alan Barich, in your absence, indicated that the crack fell within your company's quality control standards and was therefore acceptable. After expressing some doubts concerning the eddy current procedure, Alan indicated that if the crack was visible it should be polished or ground out. This was done.

As the pistons were being changed out, it was noted that some small fretting of the master connecting rods had occurred. During the course of the disassembly of the port engine, two master connecting rods and boxes were discovered to be cracked.

with cracks. As with the Star of Texas many of these cracks were severe. Also, the cracks appeared to originate from two different areas. In addition to the cracks around the boss area, other pistons were cracked along the skirt in the area of the wrist pin. Attachments No. 17 - 21 are photographs of these pistons. Are you certain that both types of cracks are due to lack of proper stress relieving? Furthermore two wrist pins were damaged (Attachment No. 21). Despite its clear relationsip to the defective pistons, Falcon was required to pay $4,000 prior to Delaval shipping two replacement wrist pins. We must again strongly object.

On December 12, 1983, as the port engine was being disassembled it was discovered that a cylinder chamber was filled with water. A dye check of the cylinder head revealed a crack in the bridge area between the exhaust valves. Again, as with the connecting rods, the cause of the crack is unclear. Even though this issue is outside the scope of this presentation, I must strongly voice my concern over this discovery. This appears to be a manufacturing problem and should be the responsibility of Delaval. Even assuming that the warranty is applicable it is unreasonable for a manufacturer to deliver a part with an inherent defect and then plead that the "warranty" limits its responsibility. Falcon should not be required, as it has been required, to pay the cost of approximately $26,000 for a replacement cylinder head.

My concerns have been heightened as the result of the piston cracks discovered on the Shoreham Nuclear Power Plant. It was my understanding that the AF pistons on this engine had been properly stress relieved. Despite this and with less than 600 hours of operating time, 23 of the 24 pistons were found to be cracked. In this case, Delaval replaced each of the AF pistons with the new AE version.

This chain of events has the tendency to cloud in our minds the continued reliability of the AN pistons presently in our engines. It is quite possible that the AE pistons may have to be eventually installed in our vessels, if the second set of "corrective" measures do not lie at the heart of the problem. In many respects, Falcon has reached the end of its capacity to finance this experimental approach to engine design.

In an attempt to discover the true source of the piston failures, our path has lead us to the experience of the MV COLUMBIA. It is my understanding that in March of this year the two Delaval Enterprise Model DMRV-16-4 units (Serial #73034 Port) (Serial #72033 Starboard) on that vessel were derated. The engines were derated from 9,200 BHP/450 ERPM to a maximum continuous rating of 6,164 BHP/403 ERPM. If true, this obviously raises doubts in my mind concerning the proper rating of our engines. As you are aware, the engines have never reached their rated capacity. Delaval's position is again to link this problem to an oversized propeller. As with the MV COLUMBIA, I now pose the question as to whether

"Delaval further warrants and represents that the engines shall be capable of extended operation while continuously burning No. two diesel fuel and/or heavy fuel oil of 4000 SSU at 100 degrees F., the chemical composition of which will be at a miniumum the degree of purity as shown on Appendix "B" attached." (emphasis added)

It is reasonable to expect that "extended operation" means a normal marine engine life. If these engines have been overrated, the increased stresses on the engine will dramatically shorten its useful life. It is quite possible that these pistons may be perfectly adequate in an engine that has less of a performance rating. In summary, the design of these engines may be falling exceedingly short of their represented performance characteristics on extended operation. This may explain the cracking of various components and the persistent engine problems.

INCREASED COSTS AND DELAYS

My discussion of the piston failures would not be complete without a review of the cost impact to Falcon. When Falcon contracted for these vessels, certain specifications were established in order to permit Falcon to predict the costs that would be incurred while operating the vessels. With relevance to the main propulsion system, the speed of the vessel, and type of fuel burned and the engine repair costs were key factors. With this information Falcon would be capable of bidding for cargoes with a reasonable expectation of the vessel's speed, fuel expenses, and engine repair costs.

We have attempted to set forth those costs that are reasonably connected to and directly related to the piston failures. We have not attempted in this presentation to charge Delaval and/or Levington with the more indirect, consequential damages suffered by Falcon. As an example, the cost for chartering the vessel in Alexandria ($150,0000) is not being claimed in this effort to settle this matter. Each of the costs are supported by our accounting records. At any time, I invite your personal review and examination of these records to satisfy yourself of the enormous out of pocket costs being expended by Falcon. Attachment No. 22 is a preliminary summary of these costs.

RECOMMENDED ACTIONS

The increasing cost for repairing the engines coupled with the down-time of the vessels has reached a point where the economic viability of our business is in serious question. Immediate steps must be taken by Falcon and Transamerica Delaval to halt this deterioration. As a first step, I believe that technical representives from both Delaval and Falcon should meet on a more regular basis. These meetings should be held whenever possible on the vessels. Each of the problems should be carefully outlined and a step by step evaluation made in an effort to correct them. The first tasks for this group should be an evaluation of the past piston failures, the procedures developed to correct such problems, a review of the replacement piston history and a design evaluation of the AN piston. While it is unfortunate

Falcon is also requesting, as a courtesy, that Delaval submit copies of all notifications it is required to give to the Nuclear Regulatory Commission. Finally, Falcon is requesting some volume discount on its purchase of spare parts. Based on the volume of spare parts being purchased by Falcon it is unreasonable that no volume discount is given. My repeated requests for this concession have to date fallen on unsympathetic ears.

CONCLUSION

Falcon has experienced many problems with the operation of the Transamerica Delaval Enterprise engines. As I noted earlier I did intend to discuss all of these problems in detail. The purpose of this presentation was to focus on the facts, cost, and our conclusions concerning the recent piston failures. We have suffered significant costs as a result of such failures. Again, our accounting records are open for your inspection to support this fact. Aside from the actual out-of-pocket expenses suffered by Falcon, we have also experienced intangible losses that cannot be quantified in dollars. The impact of the piston problem on our operating personnel has been great. Our vessels are slowly gaining the reputation of being "trouble" ships, and as a result, keeping good people is becoming a problem. Constant engine problems are also impacting on the normal maintenance of other equipment on the vessels. We are

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 2 2 1984

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MEMORANDUM FOR: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: ABNORMAL OCCURRENCE REPORT TO CONGRESS
FOR FOURTH QUARTER CY 1983

We have reviewed your March 5, 1984 memorandum containing the draft Abnormal Occurrence Report to Congress for Fourth Quarter CY 1983. The draft report contains one proposed abnormal occurrence for licensed nuclear plants. This item is:

AO 83-15 Emergency Diesel Generator Problems (a generic problem involving Transamerica Delaval, Inc., equipment)

Our comments on an earlier draft of the Fourth Quarter CY 1983 Report (reference the January 25, 1984 memorandum from H. R. Denton to C. J. Heltemes, Jr.) included a draft copy of a Commission Paper addressing the problems of TDI diesel generators. This was to be the basis for the A.O. writeup. Since that time, a number of significant changes have taken place. This has resulted in outdating the proposed A.O. writeup. Therefore we have enclosed (Enclosure 1) a proposed rewrite of the TDI diesel generator A.O. report.

In addition, our previous memorandum on the Fourth Quarter CY 1983 A.O. Report stated that NRR was investigating the consequences of Indian Point Unit 2 operating from October 24, 1983 to November 29, 1983 with the containment spray system isolation valves locked in the closed position. This valve configuration would have prevented automatic actuation of the containment spray system had it been required (a similar valve mispositioning at Farley Unit 2 resulted in Abnormal Occurrence Report #82-7).

We have completed our analyses of the Indian Point Unit 2 design basis LOCA without the mitigating affects of the containment spray system. We conclude that (1) the containment pressure would not have been exceeded but (2) the dose limitations at the Exclusion Area Boundary would have been exceeded. Therefore we have enclosed (Enclosure 2) a proposed abnormal occurrence writeup based upon major degradation of essential safety-related equipment such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could take place.

We understand that your schedule does not allow sufficient time to include the Indian Point Unit 2 event in the Fourth Quarter CY 1983 Report. Therefore, we are forwarding our writeup for your consideration as either (1) an input to the First Quarter CY 1984 Report or (2) a separate report to Congress.

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With these comments, we concur with the proposed Commission Paper.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Rewrite of TDI Diesel Generator A.O. Report
2. Proposed A.O. Writeup on Indian Point 2

ENCLOSURE 1

REPORT TO CONGRESS OF ABNORMAL OCCURRENCES

OCTOBER-DECEMBER 1983

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth calendar quarter of 1983. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

83-15 Emergency Diesel Generator Problems

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 12 of "For All Licensees") of this report notes that incidents with implications for similar facilities (generic incidents), which create major safety concerns, can be considered an abnormal occurrence. The problem discussed below involving the Transamerica Delaval, Inc. (TDI) emergency diesel generators (EDGs) at the Shoreham Nuclear Power Plant were previously described in Appendix C of NUREG-0090, Vol. 6, No. 3. It was not reported as an abnormal occurrence at that time because the immediate problem involved a plant still under construction. However, it was mentioned that reliability of the TDI EDGs remained under active review. It has now been determined that the question of reliability of TDI diesels has generic implications and should be reported as an abnormal occurrence.

Date and Place - On August 12, 1983, EDG-102 at the Shoreham Nuclear Power Plant (99% construction completion) failed due to a fractured crankshaft. The applicant for the plant is Long Island Lighting Company. The plant is a boiling water reactor and is located in Suffolk County, New York. There are three EDG units at Shoreham, all manufactured by TDI. During the following investigations of the failure and needed repairs, several conditions were identified which raised questions about the reliability of all TDI diesels at other nuclear power stations.

Nature and Probable Consequences - The failure at Shoreham occurred after 1.75 hours of testing at the two-hour overload rating (3900 kW). At the time of failure, EDG-102 had accumulated about 718 operating hours and about 19 hours at the 110% overload rating. The test in progress when the crankshaft fractured was being performed to demonstrate EDG load carrying ability following replacement of all eight cylinder heads with a newer design (originally supplied cylinder heads had developed leaks from the cooling water area).

The EDG-102 crankshaft fracture occurred on the generator (load) side of the No. 7 cylinder and extended through the load side crank arm into the crank pin. (The No. 8 cylinder is closest to the load). Examination of the other two EDGs identified cracks similar in location and orientation to the one which developed into a fracture on EDG-102. In addition, four of twenty-four connecting rod bearings were found to contain cracks in the bearing shells.

The EDGs are TDI Model DSR-48 diesels. These EDGs are the only DSR-48 diesels manufactured with a crankshaft assembly having an 11" crank pin diameter and 13" crankshaft diameter (11 x 13). On November 3, 1983, the applicant and its technical consultant reported that the crankshaft failures were definitely caused by a basic design inadequacy. Independent analysis by the contractor established that the crankshaft was overstressed relative to industry standards, a conclusion supported by various considerations, including: industry-standard torsional analysis methods, detailed stress analyses, and actual torsional test results on EDG-101. Factors contributing to the bearing cracks were found to include unsupported, overhung bearing ends, excessive crank pin journal yawing, and the presence of large pores or voids in the aluminum bearing shells.

In 1974, the applicant contracted with TDI to purchase three EDGs for the Shoreham station. This was the first order received by TDI to provide an EDG for a commercial nuclear power station. Pre-operational testing of the engines at Shoreham commenced in late 1981. Each engine has eight cylinders in a straight line (straight-8). One of the Shoreham engines had been used by TDI to qualify the straight-8 series (R48) diesel engine for nuclear service. Since testing began, the licensee has experienced several problems with the EDGs. Many component parts required reworking, redesign, and/or replacement.

At the present time, only two plants with operating licenses have TDI engines installed. One is San Onofre Unit 1 which has been shut down since February 27, 1982 for seismic modifications. The other is Grand Gulf which is authorized for power only up to 5%. A third operating plant, Rancho Seco, is presently installing TDI engines to supplement the existing non-TDI engines.

Grand Gulf has also experienced several problems with TDI engines. In 1981, preoperational testing of two V-16 engines at Grand Gulf commenced. These engines represent the first V-16 units ordered from TDI; one of the Grand Gulf engines was used to qualify the TDI V-16 line of machines for nuclear applications.

There have been a total of 57 TDI engines ordered for 16 nuclear power plant sites in the United States. A list of these sites is shown in Table 1. Only San Onofre Unit 1, Grand Gulf, and Shoreham have any significant equipment run time; therefore, the experience base of TDI units in United States nuclear service is limited.

Cause or Causes - The large number of failures together with the inspection history of TDI described below, indicate that quality assurance problems exist at TDI.

Table 1

Nuclear Plants with Transamerica Delaval, Inc.
Diesel Generators

Site	Licensee	Location	Engine Model No.
Bellefonte	Tennessee Valley Authority	Jackson County, AL	DSRV 16
Catawba	Duke Power Co.	York County, SC	DSRV 16
Comanche Peak	Texas Utilities Generating Company	Somerville County, TX	DSRV 16
Grand Gulf	Mississippi Power & Light Company	Claiborne County, MS	DSRV 16
Harris	Carolina Power & Light Co.	Wake & Chatham Counties, NC	DSRV 16
Hartsville*	Tennessee Valley Authority	Trousdale & Smith Counties, TN	DSRV 16
Midland	Consumers Power Co.	Midland County, MI	DSRV 12
Perry	Cleveland Electric Illuminating Co.	Lake County, OH	DSRV 16
Phipps Bend*	Tennessee Valley Authority	Hawkins County, TN	DSRV 16
Rancho Seco	Sacramento Municipal Utility District	Sacramento County, CA	DSR 48
River Bend**	Gulf States Utilities	West Feliciana Parish, LA	DSR 48
San Onofre	Southern California Edison Co.	San Diego County, CA	DSRV 20
Shoreham	Long Island Lighting Co.	Suffolk County, CA	DSR 48
Vogtle	Georgia Power Co.	Burke County, GA	DSRV 16
WPPSS	Washington Public Power Supply System	Benton County, WA	DSRV 16
WPSS 4*	Washington Public Power Supply System	Benton County, WA	DSRV 16

*Project delayed or cancelled

**River Bend Unit 2 has been cancelled

Note: Of the plants listed above, only San Onofre Unit 1, Rancho Seco, and Grand Gulf have received operating licenses.

Actions Taken to Prevent Recurrence

Long Island Lighting Company - The applicant has replaced the three 11 x 13 crankshaft assemblies with the 12 x 13 crankshaft assemblies like these reportedly installed in all other DSR-48 diesels. In addition, the connecting rod bearings were replaced with bearings designed to accomodate the new 12" crank pin diameter and to address the factors which caused the earlier bearings to develop cracks.

The applicant still intends to license the Shoreham facility with the TDI diesel generators. However, as part of a long-term solution for the TDI diesel problems, the applicant has recently placed purchase orders for three diesel generators from Colt Industries. We understand that the applicant intends to ultimately replace the TDI diesels with Colt diesels. Delivery of the Colt diesels is scheduled for the fall of 1985 which coincides with the completion of a new diesel generator building that is currently under construction.

Additional actions will also be required in conjunction with the other actions described below.

Other Licensees - By letter dated December 23, 1983, the NRC staff was informed that a TDI diesel engine owners group had been formed to address the EDG reliability issue.

NRC - The staff is continuing to gather information regarding problems concerning TDI units, reviewing specifics of the problems, and developing a course of action to assure that the affected plants have reliable EDG capability.

On August 30, 1983, the NRC issued Inspection and Enforcement Information Notice No. 83-58 to licensees to inform them of the Shoreham event (Ref. 1). Prior to the Shoreham event, the NRC issued Information Notice No. 83-51 to licensees to inform them of various diesel generator problems (Ref. 2).

The NRC Region IV Vendor Inspection Branch performed inspections of the TDI facility in Oakland, California during July, September, and October 1983. These inspections were performed at the request of Region I (Region I has responsibility for inspection activities at Shoreham facility) and in response to allegations of irregularities in the quality assurance (QA) program. Several potential nonconformances with NRC requirements were found during the July 1983 inspections. During the September and October 1983 inspections, the staff identified conditions which indicate that portions of the TDI Quality Assurance (QA) Program may not have been carried out in accordance with the provisions of 10 CFR 50, Appendix B.

The staff has met with the applicant for Shoreham and Grand Gulf to discuss the failures to date, the results of the Shoreham investigation, and the actions to be taken to recover from the failures. The staff has also developed several lists of questions that it feels need to be addressed as part of the TDI engine evaluations. One list, which has been sent to all TDI diesel owners, requested specific information about each machine. Another was sent to TDI on December 1, 1983, requesting information about the design development history of various parts of TDI machines. Delaval responded on December 16, 1983.

On January 16, 1984, a special NRC project group was formed to coordinate the overall NRC review of TDI diesel generators. Their primary responsibility is to evaluate the overall qualification of TDI diesel generators for nuclear service. Pacific Northwest Laboratory has been chosen to assist the staff in assessing and evaluating the corrective action plans being submitted by utilities possessing TDI diesel generators.

The NRC staff held a meeting on January 26, 1984 with senior utility executives representing each of the applicants listed in Table 1. The staff informed them of its concerns regarding the breakdown in quality assurance in the TDI manufacturing facility and emphasized the significance of the widespread operating problems to date with TDI engines.

The NRC staff has concluded that before additional licensing action is taken to authorize the operation of a nuclear power plant with TDI engines, these issues, relating to quality assurance, operating experience, and the ability of the machines to reliably perform their intended function, must be addressed.

Further reports will be made as appropriate.

ENCLOSURE 2

PROPOSED ABNORMAL OCCURRENCE

83- Inoperable Containment Spray System at Indian Point Unit 2

Appendix A (see general criterion 2) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence. In addition, Example 3 under "For Commerical Nuclear Power Plants" of Appendix A notes that loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident can be considered an abnormal occurrence.

Date and Place

On November 29, 1983, while performing a bimonthly (every two months) containment spray pump surveillance test, during normal operation, two motor operated spray header discharge valves (MOV 869A and MOV 869B) were found in the locked closed, de-energized position instead of the required locked open, de-energized position. This condition would have prevented automatic actuation of the containment spray system during the safety injection phase of an accident.

Cause or Causes

A review of conditions leading up to this event revealed that on October 12, 1983, during a cold shutdown, MOVs 869A and 869B were closed and tagged out of service to work on the reactor coolant system. On October 18, 1983, while still in the cold shutdown condition, the tagout was cleared; however, these valves were specified to remain closed to block the containment spray paths while personnel continued to work in the containment. On October 28, 1983, prior to plant startup, operators were assigned to perform a Safety Injection System Check-Off List (COL-12) check-off which should have returned MOV 869A and MOV 869B to their proper positions prior to heating the reactor coolant system above 350 degrees. However, the personnel who conducted this check-off did not verify the positions of MOV 869A and 869B.

Upon discovery by plant personnel, the incident was immediately reported by telephone to the NRC Operations Center. An investigation was initiated to establish the cause of the event and recommend corrective action. The investigation included interviews with cognizant operations and test personnel, a review of the COLs (Check-Off Lists), OADs (Operation Administrative Directives), Training and Operator Qualification Program, the facility Technical Specifications, FSAR, Indian Point Probabilistic Safety Study, NRC's Safety Evaluation Report and other reference documentation.

COL-12 was performed on October 23 and 24, 1983. It required one operator to ensure the correct valve position and a second operator to verify the position. COL-12 directs the operators to the motor control centers to perform two verifications for each valve: (1) verify that the position of the valve is open, and (2) verify that the breaker is de-energized. In the de-energized condition, position indication for the valve is lost at the motor control centers. Verifying position at the motor control center, therefore, requires energizing the breaker. The first operator assumed that the valve was positioned by another operator. The second operator assumed the valve was open because the breaker was locked in the de-energized position.

Test personnel described how they found the valves and their subsequent actions. Test personnel realized the valve line-up was wrong when the "as left" position differed from the "as found" position during the spray pump test. The SRO was notified when the discrepancy was identified and the valves were positioned correctly.

As a result of the investigation, it was determined that improvements could be made in the training/qualification program of a nuclear plant operator to place new emphasis on equipment status identification. The operator qualification standard will specify the knowledge required by the operator for the performance of COLs. In addition, the licensee will further assure that appropriate guidance is provided to the operators in the conduct of COLs.

Nature and Probable Consequences

The Unit No. 2 FSAR Sections 6.4 and 14.3 present the original analyses for the facility showing that the containment air recirculation cooling and filtration system will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, and has sufficient filtration capacity to reduce the concentration of fission products in the containment atmosphere following a loss of reactor coolant to levels ensuring that the two-hour and the thirty-day thyroid doses will not exceed the guideline limits of 10 CFR Part 100.

The containment heat removal system consists of five containment fan cooler units and two containment spray trains. The Unit No. 2 FSAR states that sufficient post-accident heat removal capability can be provided by any of the following combinations:

1. All five containment fan cooling units;

2. Both containment spray trains (and one of the two recirculation spray trains); or

3. Three containment fan cooler units and one containment spray train.

In addition to their cooling capability, the containment fan cooling units also provide filtration of the post-accident atmosphere. This is accomplished by directing the fan's discharge through a filtration system.

During the time in question, automatic actuation of the containment spray system would not have been possible. However, the licensee has stated that several indications would have been available in the control room to inform the reactor operators that spray injection was not taking place. These include:

1. A direct indication would be lack of flow indicated on FI-930, Spray Additive Tank Discharge.

2. A second indication would be the rate of decrease of the Refueling Water Storage Tank level during the five (5) minute intervals when the operator is required to check tank level in accordance with procedure E-1.

3. A third indication would be the failure of the Spray Additive Tank (SAT) level to fall and not receiving the expected SAT Low Level Alarm.

With several possible indications of a problem, the operator could notice the lack of spray flow from immediately after spray initiation up to thirty minutes after initiation. There are a number of options available to the reactor operators.

1. Valve realignment could take place quickly from MCC26AA and MCC26BB. The MCC area is designed to be accessible in high post-accident radiation fields.

2. Spray could be supplied from the RHR Pump discharge by opening MOV 889A or MOV 889B from the Central Control Room.

Although the reactor operators would be expected to recognize that the containment spray isolation valves were closed in a timely manner, the NRC staff has peformed bounding calculations to predict worst case conditions. Reanalysis of the design basis accident were performed in order to verify that (1) the containment design pressure was not exceeded and (2) the post-accident off-site dose limitations were not exceeded.

Indian Point Unit 2 has two trains of fan coolers on separate power sources; one train has two fan coolers and the other train has three fan coolers. Since, for the present situation, both containment spray trains would be out of service, the staff assumed a single active failure would reduce the active containment heat removal capability to two fan coolers during a pipe break accident. As previously mentioned, this assumption will increase the offsite dose calculations because of the filtration that is performed by the fan cooler units.

Pressure Response

The peak calculated containment pressure for the design basis LOCA (double-ended pump suction guillotine break) is 41.9 psig. This is slightly above the licensee's previously calculated peak pressure of 40 psig (with containment sprays) but substantially below the containment design pressure of 47 psig.

Based on our evaluation, we conclude that the containment design pressure would not have been exceeded and containment integrity would not have been jeopardized had an accident occurred with the containment spray system inoperable.

Off-site Dose Responses

The methods and assumptions used by the staff in this evaluation were consistent with those used in the current licensee application reviews (i.e., Standard Review Plan 15.6.5, "Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary"). Our evaluation concludes that the resultant doses would be approximately four times the 10 CFR Part 100 thyroid exposure guidelines at the Exclusion Area Boundary.

An additional analysis was performed assuming that operator action was taken taken to initiate containment spray after 30 minutes. This calculation indicates that the resultant dose would be approximately 1.8 times the exposure guidelines at the Exclusion Area Boundary.

Possible Mitigating Factors

The above dose calculations assumed the standard containment leak rate of 0.1% for the first 24 hours. Credit for a reduced leak rate has not been given for either (1) the actual, as measured containment leak rate or (2) the Isolation Valve Seal Water System which automatically injects water between the containment isolation valves post-accident in order to eliminate potential containment leak paths.

The calculations assume the worst case single active failure (i.e., the power source that powers three of the five containments fan cooler units). In addition, credit is not given to operator action to actuate the containment spray systems prior to 30 minutes.

Actions Taken to Prevent Recurrence

By the Licensee - The event has been attributed to personnel error by the valve positioner and the valve position checker. Among immediate corrective action steps taken by the licensee included verifying correct valve positions of similarly de-energized safeguards valves found on check-off lists.

Among long term corrective action steps are on number of improvements in the licensee's quality assurance and training programs. In addition:

1. A review of valve position indication for all safety related valves will be made to determine if modifications are necessary to provide for positive indication of de-energized valves.

2. The operability of all currently installed safety related MOV position indicators will be verified and corrected if necessary.

By the NRC - The staff considers this an isolated event. Other than monitoring the licensee's activities, the staff has no followup plans regarding this event.

An enforcement conference was held in the NRC Region I (King of Prussia, Pennsylvania) office with the licensee on December 13, 1983. The licensee presented their program for preventing recurrence. The NRC concurred with the licensee's corrective actions.

NRC Region I performed inspections to determine the circumstances associated with this event. (AEOD to supply information on Enforcement actions).

This incident is closed for purposes of this report.

NIRS

Nuclear Information and Resource Service
1346 Connecticut Avenue NW, 4th Floor, Washington, D.C. 20036 (202) 296-7552

April 10, 1984

Director
Office of Adminstration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FREEDOM OF INFORMATION ACT REQUEST
FOIA-84-275
Rec'd 4-16-84

FREEDOM OF INFORMATION ACT REQUEST

To whom it may concern:

Pursuant to the Freedom of Information Act, 5 U.S.C. 522, as amended, the Nuclear Information and Resource Service requests the following documents regarding Transamerica Delaval Inc. (TDI) diesel generators installed at the Shearon Harris nuclear plant. Please consider "documents" to include reports, studies, test results, correspondence, memoranda, meeting notes,meeting minutes, working papers, graphs, charts, diagrams, notes and summaries of conversations and interviews, computer records, and any other forms of written communication, including internal NRC Staff memoranda. The documents are specifically requested from, but not limited to, the Office of Inspection and Enforcement (I&E); Office of the Executive Legal Director (OELD); Office of Analysis and Evaluation of Operational Data (AEOD); Office of Nuclear Regulatory Research (Research); Office of Nuclear Reactor Regulation (NRR); and the Operating Reactors Branches of the Division of Licensing. In your response, please identify which documents correspond to which requests below.

Pursuant to this request, please provide all documents prepared or utilized by, in the possession of, or routed through the NRC related to:

1. The TDI diesel generators at the Shearon Harris nuclear plant; and

2. All lists of problems and defects which have occured with TDI generators being used or tested, or which have not yet been used, for nuclear facilities and in other applications (e.g. marine).

In our opinion, it is appropriate in this case for you to waive copying and search charges, pursuant to 5 U.S.C. 552(a)(4)(A) "because furnishing the information can be considered as primarily benefiting the general public." The

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Nuclear Information and Resource Service is a non-profit organization serving local organizations concerned about nuclear power and providing information to the general public.

Sincerely,

Nina Bell

Nina Bell
Nuclear Safety Analyst

cc: File