NRC FORM 366

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 N J O C P 1 (2) 0 0 - 0 0 0 0 0 - 0 0 (3) 4 1 1 1 1 (4) (5)
LICENSEE CODE / LICENSE NUMBER / LICENSE TYPE / CAT

CON'T 01 REPORT SOURCE L (6) 0 5 0 0 0 2 1 9 (7) 0 2 0 2 8 1 (8) 0 4 0 2 8 1 (9)
DOCKET NUMBER / EVENT DATE / REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

During routine surveillance testing main steamline high radiation monitor RN-06B tripped at a value higher than that specified in T.S. Table 3.1.1 (less than or equal to ten times normal background radiation levels at full rated power). The other three channels were operating within desired limits and would have initiated the required protective actions. Monitor RN-06B would also have functioned but at a slightly higher level. There were no effects on the public health or safety.

SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
I A (11)	X (12)	X (13)	I N S T R U (14)	I (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 1	—	0 1 0	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
E (18)	X (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	Y (24)	A (25)	G 0 8 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

The monitor was operating well within its design accuracy. The cause is attributed to a failure to revise the trip settings due to changes in the normal background radiation levels at full rated power. The alarm and trip settings will be lowered to be more conservative.

FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
E (28)	0 9 0 (29)	NA	B (31)	Surveillance Test

ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

NUMBER	TYPE	DESCRIPTION (39)
0 0 0 (37)	Z (38)	NA

8104070568

PERSONNEL INJURIES

NUMBER	DESCRIPTION (41)
0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

TYPE	DESCRIPTION
Z (42)	NA

PDR

PUBLICITY (45)

ISSUED	DESCRIPTION
Y (44)	Weekly News Release

NRC USE ONLY

SSINS No.: 6820
Accession No.:
8011040289
IEB 81-03

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 10, 1981

IE Bulletin 81-03 : FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS BY CORBICULA SP. (ASIATIC CLAM) AND MYTILUS SP. (MUSSEL)

Description of Circumstances:

On September 3, 1980, Arkansas Nuclear One (ANO), Unit 2, was shut down after the NRC Resident Inspector discovered that Unit 2 had failed to meet the technical specification requirements for minimum service water flow rate through the containment cooling units (CCUs). After plant shutdown, Arkansas Power and Light Company, the licensee, determined that the inadequate flow was due to extensive plugging of the CCUs by Asiatic clams (Corbicula species, a non-native fresh water bivalve mollusk). The licensee disassembled the service water piping at the coolers. Clams were found in the 3-inch diameter supply piping at the inlet to the CCUs and in the cooler inlet water boxes. Some of the clams found were alive, but most of the debris consisted of shells. The size of the clams varied from the larvae stage up to one inch. The service water, which is taken from the Dardanelle Reservoir, is filtered before it is pumped through the system. The strainers on the service water pump discharges were examined and found to be intact. Since these strainers have a 3/16-inch mesh, much smaller than some of the shells found, it appears that clams had been growing in the system.

Following the discovery of Asiatic clams in the containment coolers of Unit 2, the licensee examined other equipment cooled by service water in both Units 1 and 2. Inspection of other heat exchangers in the Unit 2 service water system revealed some fouling or plugging of additional coolers (seal water coolers for both redundant containment spray pumps and one low-pressure safety injection pump) due to a buildup of silt, corrosion products, and debris (mostly clam shell pieces). The high-pressure safety injection (HPSI) pump bearing and seal coolers were found to have substantial plugging in the 1/2-inch pipe service water supply lines. The plugging resulted from an accumulation of silt and corrosion products.

Clam shells were found in some auxiliary building room coolers and in the auxiliary cooling water system which serves non-safety-related equipment.

Flow rates measured during surveillance testing through the CCUs at ANO-2 had deteriorated over a number of months. Flushing after plant shutdown initially resulted in a further reduction in flow. Proper flow rates were restored only after the clam debris had been removed manually from the CCUs.

The examination of the Unit 1 service water system revealed that the "C" and "D" containment coolers were clogged by clams. Clams were found in the 3-inch inlet headers and in the inlet water boxes. However, no clams were found

8011040289

in the "A" and "B" coolers. This fouling was not discovered during surveillance testing because there was no flow instrumentation on these coolers.

The service water system in Unit 1 was not fouled other than stated above, and the licensee attributed this to the fact that the service water pump suctions are located behind the main condenser circulating pumps in the intake structure. It was thought that silt and clams entering the intake bays would be swept through the condenser by the main circulating pumps and would not accumulate in the back of the intake bays. In contrast, Unit 2 has no main circulating pumps in its intake structure because condenser heat is rejected through a cooling tower via a closed cooling system. As a result of lower flowrates of water through the Unit 2 intake structure, silt and clams could have a tendency to accumulate more rapidly in Unit 2 than in Unit 1. During the September outage, clams and shells were found to have accumulated to depths of 3 to 4-1/2 feet in certain areas of the intake bays for Unit 2.

The Asiatic clam was first found in the United States in 1938 in the Columbia River near Knappton, Washington. Since then, Corbicula sp. has spread across the country and is now reported in at least 33 states. The Tennessee Valley Authority (TVA) power plants also have experienced fouling caused by these clams. They were first found in the condensers and service water systems at the Shawnee Steam Plant in 1957. Asiatic clams were later found in the Browns Ferry Nuclear Plant in October 1974 only a few months after it went into operation. This initial clam infestation at Browns Ferry was enhanced by the fact that, during the final stages of construction, the cooling water systems were allowed to remain filled with water for long periods of time while the systems were not in use. This condition was conducive to the growth and accumulation of clams. Since that time, the Asiatic clam has spread across the Tennessee Valley region and is found at virtually all the TVA steam-electric and hydroelectric generating stations.

Present control procedures for Asiatic clams vary from station to station and in their degree of effectiveness. The use of shock chlorination during surveillance testing as the only method of controlling biofouling by this organism appears to be ineffective. The level of fouling has been reduced to acceptable levels at TVA stations by using continuous chlorination during peak spawning periods, clam traps, and mechanical cleaning during station outages.

The results of a series of tests on mollusks performed at the Savannah River facility showed that mature Corbicula sp. had as much as a 10 percent survival rate after being exposed to high concentrations of free residual chlorine (10 to 40 ppm) for up to 54 hours. When the clams were allowed to remain buried in a couple of inches of mud, their survival rates were as high as 65 percent.

In studies on shelled larvae, approximately 200 microns in size, TVA reported preliminary results indicating that a total chlorine residual of 0.30 to 0.40 ppm for 96 to 108 hours would be required to achieve 100 percent control of the Asiatic clam larvae.

Corbicula sp. has also shown an amazing ability to survive even when removed from the water. Average times to death when left in the air have been reported for low relative humidity as 6.7 days at 30°C (86°F) and 13.9 days at 20°C (68°F) and for high relative humidity as 8.3 days at 30°C and 26.8 days at 20°C.

Corbicula sp. on the other hand, has shown a much greater sensitivity to heat. Tests performed by TVA resulted in 100 percent mortality of clam larvae, very young clams, and 2mm clams when they were exposed to 47°C (117°F) water for 2 minutes. Mature clams, up to 14mm, were also tested and all died at 47°C following a 2 minute exposure. A statistical analysis of the 2 minute exposure test data revealed that a temperature of 49°C (120°F) was necessary to reach the 99 percent confidence level of mortality for clams of the size tested.

To date, heat has been shown to be the most effective way of producing 100 percent mortality for the Asiatic clam. At ANO, the service water system was flushed with 77°C (170°F) water obtained from the auxiliary boiler for approximately one half hour; 100 percent mortality was expected.

A similar problem has occurred with mussels (*Mytilus* sp.). Infestations of mussels have caused flow blockage of cooling water to safety-related equipment at nuclear plants such as Pilgrim and Millstone. Unlike the Asiatic clam, mussels cause biofouling in salt water cooling systems.

The event at ANO is significant to reactor safety because (1) the fouling represented an actual common cause failure, i.e. inability of safety system redundant components to perform their intended safety functions, and (2) the licensee was not aware that safety system components were fouled. Although the fouling at ANO-2 developed over a number of months, neither the licensee management control system nor periodic maintenance or surveillance program detected the failure.

ACTIONS TO BE TAKEN BY LICENSEES

Holders of Operating Licenses:

1. Determine whether *Corbicula* sp. or *Mytilus* sp. is present in the vicinity of the station (local environment) in either the source or receiving water body. If the results of current field monitoring programs provide reasonable evidence that neither of these species is present in the local environment, no further action is necessary except for items 4 and 5 in this section for holders of operating licenses.

2. If it is unknown whether either of these species is present in the local environment or is confirmed that either is present, determine whether fire protection or safety-related systems that directly circulate water from the station source or receiving water body are fouled by clams or mussels or debris consisting of their shells. An acceptable method of confirming the absence of organisms or shell debris consists of opening and visually examining a representative sample of components in potentially affected safety systems and a sample of locations in potentially affected

fire protection systems. The sample shall have included a distribution of components with supply and return piping of various diameters which exist in the potentially affected systems. This inspection shall have been conducted since the last clam or mussel spawning season or within the nine month period preceding the date of this bulletin. If the absence of organisms or shell debris has been confirmed by such an inspection or another method which the licensee shall describe in the response (subject to NRC evaluation and acceptance), no further action is necessary except for items 4 and 5 of actions applicable to holders of an operating license.

3. If clams, mussels or shells were found in potentially affected systems or their absence was not confirmed by action in item 2 above, measure the flow rates through individual components in potentially affected systems to confirm adequate flow rates i.e., flow blockage or degradation to an unacceptably low flow rate has not occurred. To be acceptable for this determination, these measurements shall have been made within six months of the date of this bulletin using calibrated flow instruments. Differential pressure (DP) measurements between supply and return lines for an individual component and DP or flow measurements for parallel connected individual coolers or components are not acceptable if flow blockage or degradation could cause the observed DP or be masked in parallel flow paths.

   Other methods may be used which give conclusive evidence that flow blockage or degradation to unacceptably low flow rates has not occurred. If another method is used, the basis of its acceptance for this determination shall be included in the response to this bulletin.

   If the above flow rates cannot be measured or indicate significant flow degradation, potentially affected systems shall be inspected according to item 2 above or by an acceptable alternative method and cleaned as necessary. This action shall be taken within the time period prescribed for submittal of the report to NRC.

4. Describe methods either in use or planned (including implementation date) for preventing and detecting future flow blockage or degradation due to clams or mussels or shell debris. Include the following information in this description:

   a. Evaluation of the potential for intrusion of the organisms into these systems due to low water level and high velocities in the intake structure expected during worst case conditions.

   b. Evaluation of effectiveness of prevention and detection methods used in the past or present or planned for future use.

5. Describe the actions taken in items 1 through 3 above and include the following information:

   a. Applicable portions of the environmental monitoring program including last sample date and results.

b. Components and systems affected.

c. Extent of fouling if any existed.

d. How and when fouling was discovered.

e. Corrective and preventive actions.

Holders of Construction Permits:

1. Determine whether Corbicula sp. or Mytilus sp. is present in the vicinity of the station by completing items 1 and 4 above that apply to operating licenses (OL).

2. If these organisms are present in the local environment and potentially affected systems have been filled from the station source or receiving water body, determine whether infestation has occurred.

3. Describe the actions taken in items 1 and 2 above for construction permit holders and include the following information:

   a. Applicable portions of the environmental monitoring program including last sample date and results.

   b. Components and systems affected.

   c. Extent of fouling if any existed.

   d. How and when fouling was discovered.

   e. Corrective and preventive actions.

Licensees of facilities with operating licenses shall provide the requested report within 45 days of the date of this bulletin. Licensees of facilities with construction permits shall provide the report within 90 days.

Provide written reports as required above, signed under oath or affirmation, under the provisions of Section 182a of the Atomic Energy Act of 1954. Reports shall be submitted to the Director of the appropriate Regional Office and a copy forwarded to the Director, Office of Inspection and Enforcement, NRC, Washington, D.C. 20555.

This request for information was approved by GAO under a blanket clearance number R0072 which expires November 30, 1983. Comments on burden and duplication should be directed to Office of Management and Budget, Room 3201, New Executive Office Building, Washington, D.C. 20503.

Attachment:
Recently issued IE Bulletins

Attachment
IEB 81-03
April 10, 1981

RECENTLY ISSUED
IE BULLETINS

Bulletin No.	Subject	Date Issued	Issued To
81-02	Failure of Gate Type Valves to Close Against Differential Pressure	3/9/81	All power reactor facilities with an OL or CP
81-01 Rev. 1	Surveillance of Mechanical Snubbers	3/5/81	All power reactor facilities with an OL & specified facilities with CP
80-17, Supplement 5	Failure of Control Rods to Insert During a Scram	2/13/81	All BWR facilities with OL or CP
81-01	Surveillance of Mechanical Snubbers	1/27/81	All power reactor facilities with OL & to specified facilities with CP
80-25	Operating Problems with Target Rock Safety-Relief Valves at BWRs	12/19/80	All BWR facilities with OL & specified near term OL BWR facilities & all BWRs with a CP
Supplement 4 to 80-17	Failure of Control Rods to Insert During a Scram at a BWR	12/18/80	To specified BWRs with an OL & All BWRs with a CP
80-24	Prevention of Damage Due to Water Leakage Inside Containment (October 17, 1980 Indian Point 2 Event)	11/21/80	All power reactor facilities with OL or CP
80-23	Failures of Solenoid Valves Manufactured by Valcor Engineering Corporation	11/14/80	All power reactor facilities with OL or CP
80-22	Automation Industries, Model 200-520-008 Sealed-Source Connectors	9/11/80	All radiography licensees

OL = Operating License
CP = Construction Permit

Date: April 21, 198

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-I-81-61

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Northeast Nuclear Energy Company
Millstone Nuclear Station, Unit 1
(DN 50-245)
Waterford, CT

Subject: TURBINE FAILURE

At 2:26 a.m., April 21, the main turbine was tripped due to receipt of high vibration indication. The reactor was at 30% power with 200 Mw(e) output from the turbine generator. High conductivity in the condenser hotwell (150 micromhos/cm), six minutes after the turbine trip, indicated that a missile (blading or shrouding) from the turbine had caused the failure of a number of condenser tubes. The reactor was scrammed manually at 2:32 a.m. and isolated by shutting the main steam isolation valves (MSIV's). The isolation condenser was not available due to low water level and the reactor was depressurized through manual operation of one safety relief valve. Normal reactor water level was maintained by the feedwater system; condensate demineralizer effluent conductivity remained at 0.06 micromhos/cm, with no chlorides indicated. In anticipation of chloride breakthrough of the demineralizers which would have rendered the normal feedwater system unuseable for reactor feed, all LPCI and core spray pumps (low pressure ECCS) were started manually but injection valves were kept closed. Satisfactory feedwater conductivity was maintained during the subsequent reactor cooldown and the low pressure ECCS systems were not needed (injection valves were not opened). Peak torus temperature was 104° F. The shutdown cooling system was placed in service at 4:00 a.m.

The reactor is presently at 250° F and vented. Peak recorded reactor water conductivity was 0.1 micromhos/cm. The containment atmosphere is 1×10^{-9} microcuries/cc. No radioactive releases occurred.

The plant is expected to be out of service for some time. Media interest is expected. The licensee has issued a press release. NRC will respond to media inquiries. The State of Connecticut has been informed. The licensee notified the NRC of this incident via the ENS at 2:34 a.m. The resident inspectors are following up on this event. This PN is issued for information only.

~~8104280243~~

Contact:	J. SHEDLOSKY	J. McCANN	R. KEIMIG
	842-7087	488-1235	488-1224

263

RIBUTION: (Facsimile Transmission Times Noted)

BLDG 10:48
rman Ahearne
issioner Gilinsky
issioner Bradford
issioner Hendrie

Chilk, SECY
Kammerer, CA

LIPS BLDG 10:32
Denton, NRR
Vollmer, NRR
Murley, NRR
Ross, NRR
Eisenhut, NRR
Hanauer, NRR

MARYLAND NATIONAL
BANK BUILDING 10:51
W. J. Dircks, EDO
C. Michelson, AEOD
J. J. Fouchard, PA
N. M. Haller, MPA
G. W. Kerr, OSP
H. K. Shapar, ELD

WILLSTE BLDG 12:40
J. G. Davis, NMSS
R. B. Minogue, RES

LANDOW BLDG (Fax by HQ)
(6 MIN/PAGE)
J. J. Cummings, OIA

IE:TAS 10:45
NRC/TMI Program
Office 1:55

Region II 10:54
Region III 11:15
Region IV 2:15
Region V 2:30

RI Resident Sites
Millstone / 1:01
/

(MAIL)
R. G. Smith, SD
IE:TAS
Document Mgt. Br.
S. Ebbin, NSOC

DMB

TOLEDO
EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346
License No. NPF-3
Serial No. 1-201
May 22, 1981

RECEIVED
JUL 2 0 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. James G. Keppler
Regional Director, Region III
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

IE Bulletin No. 79-14, dated July 2, 1979, requested that we develop and implement an inspection program to verify that the Davis-Besse Nuclear Power Station, Unit No. 1, seismic analysis input of safety related piping systems conforms to the actual field conditions.

On June 30, 1980, we reported to you the results of our detailed engineering reviews for normally inaccessible safety related piping. As part of that submittal, we transmitted our schedule for follow-on analytical work required under Item No. 4B of the Bulletin. On February 13, 1981, we submitted a revised schedule for the above follow-on analytical work. Attached is a description of the results of our follow-on analysis of the identified discrepancies for normally inaccessible safety related piping in accordance with Item No. 4B of the Bulletin.

Yours very truly,

R.P. Crouse/nak

RPC:CLM

Attachment

bj c/5

cc
U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operation Inspection
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Operating Reactors
Washington, D.C. 20555

NRC Davis-Besse Resident Inspector

PDR 8106190725

IE11
S 1/1

125

Docket No. 50-346
License No. NPF-3
Serial No. 1-201
May 22, 1981

Seismic Analysis For As-Built
Safety Related Piping Systems

Response to NRC IE Bulletin No. 79-14

Davis-Besse Nuclear Power Station Unit 1

I. INTRODUCTION

NRC IE Bulletin 79-14, dated July 2, 1979, Revision 1, dated July 18, 1979, Supplement 1, dated August 15, 1979, and Supplement 2, dated September 7, 1979, required all power reactor facility licensees to verify that the seismic analysis of safety-related piping systems applies to the actual as-built configuration of systems. The action items identified in the Bulletin apply to all safety-related piping, 2-1/2 inches in diameter and greater, and to Seismic Category I piping, regardless of size, which was analyzed by computer.

The response to Item 1 of the Bulletin was submitted on August 1, 1979 (Serial No. 1-81). A response to Item 2 of the Bulletin was submitted on October 1, 1979 and October 19, 1979 (Serial Nos. 1-93 and 1-95). A partial response to Item 4 covering the normally accessible piping was submitted on June 16, 1980 (Serial No. 1-137). A complete response to Item 3 was submitted on June 30, 1980 (Serial No. 1-146).

This report provides the final response to Item 4 of the Bulletin, describing the results of the normally inaccessible piping stress reanalysis incorporating the as-built configuration of the piping and support systems.

II. Review of Inspection and Results

Inspection of all normally inaccessible safety-related piping and the remaining normally accessible safety-related piping not covered by our responses dated October 1, 1979 and October 19, 1979, due to operating conditions, began on April 14, 1980 and was completed on May 21, 1980. Clearances for all whip restraints on inaccessible piping were checked during the 1980 refueling outage and have been found to have no affect on the stress analysis. Preliminary evaluation and detailed engineering reviews were completed in accordance with Supplements 1 and 2 of the Bulletin. Discrepancies identified by both the inspection team and the stress analyst were tabulated and shown in Attachment 1 to the June 30, 1980 submittal. Evaluation of the preliminary two day reviews indicated that none of the noted nonconformances would adversely affect system operability. The detailed engineering review of all discrepancies, both those identified in the field and any subsequently identified in the office, indicated two deviations which caused an overstressed condition in the piping system and for which system operability may have been affected. Since additional stress analyses beyond the 30 day evaluation were required to determine the affect on system operability and the unit was in a refueling outage, these items were corrected prior to restart of the unit. These two deviations were summarized in the June 30, 1980 response.

III. Description of Stress Reanalysis and Results

In Attachment 1 to the June 30, 1980 submittal, it was anticipated that 44 of the 139 original stress calculations would require complete computer reanalysis. An additional 61 of the 139 stress calculations were to require a simple hand calculation to resolve the discrepancies, and 34 calculations did not require any piping reanalysis. During the course of the reanalysis effort, a number of stress calculations were re-evaluated by the stress analyst based on the results of the calculations performed thus far and it was determined that a more extensive computer analysis would be required. As a result, the reanalysis consisted of a total of 76 stress calculations which were completely computer reanalyzed, an additional 33 stress calculations which were resolved by means of a simple hand calculation, and 30 calculations which required no reanalysis.

None of the 109 stress calculations reanalyzed required the rerouting of any piping. A few of the reanalyzed stress calculations did require the addition or relocation of a support. However, the vast majority of the stress calculations, that were reanalyzed, only revised the calculated load transmitted to the existing supports.

IV. Support Reanalysis and Modifications

Pipe supports/anchors on the inaccessible safety-related piping, as defined by the bulletin, were reanalyzed for two different reasons. The piping system stress calculation reanalysis generated revised support loads that were higher than the original design loads or the inspection identified discrepancies that existed between the design drawings and the as-built configuration. In the first case, the supports were reanalyzed for the higher loads and, in the latter, a reanalysis was performed to verify the adequacy of the support.

Both of these cases combined have generated structural reanalysis for a total of approximately 300 supports and anchors out of the total of 1500 on the inaccessible piping. Of this 300, approximately 45 supports/anchors have been identified as requiring some modification to the structure to either return it to its design condition or to modify it to accommodate its new loading condition.

These modifications can be classified into the same three categories that were used to describe the modification for the accessible area supports/anchors:

1. a minor revision
2. a moderate change or addition, and
3. a major structural rework or complete redesign of the support.

Typically, minor revisions consist of the resetting of a spring or the addition of a small stiffener or shim plate. Such minor modifications comprise approximately 42 percent of the total number of support modifications for the inaccessible areas.

The moderate change or addition includes, for example, the replacement of a structural member or the addition of a brace or kicker to the support structure. Modifications in this category would not require the complete dismantling of the support but would rather affect only a portion of the structure. Approximately 56 percent of the support modifications fall in this category.

The addition of a pipe support/anchor or the redesign or relocation of the entire support is considered a major modification. Only one of the forty-five modifications falls into this category.

V. Schedule for Completion of Support Modifications

As stated previously, approximately 45 pipe supports/anchors have been identified as requiring some modification to the structure. These design modifications are currently in some stage of the engineering design cycle. The work packages are being given expeditious treatment and the projected completion date for issue of the last package is July 1, 1981.

Since these support modifications are located in normally inaccessible areas of the station, work will be performed during the next scheduled refueling outage.

VI. Schedule for Issue of As-Built Drawings

The current schedule for completion of as-built drawings incorporating the inspection findings for piping supports/anchors located on inaccessible safety-related piping is December 1, 1981.

As-built drawings reflecting the forty-five modifications performed during the refueling outage will be issued by December 1, 1982.

VII. Conclusions

IE Bulletin 79-14 inspection of the normally inaccessible piping did uncover minor discrepancies between the design and the as-built configuration of the piping and supporting systems. The affect of these discrepancies has been evaluated in detail and the preliminary conclusions made in our June 30, 1980 response are still valid, that no deficiency has been discovered that would have adversely affected the operability of any safety-related system.

Central Files

NORTHEAST UTILITIES

P.O. BOX 270
HARTFORD, CONNECTICUT 06101

04-21-81

Mr. Boyce H. Grier
Director, Region I
Office of Inspection and Enforcement
U. S. Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

RECEIVED
MAY 04 1981
U.S. NUCLEAR REGULATORY COMMISSION

Reference: Provisional License DPR-21
Docket Number 50-245
Reportable Occurrence RO-81-04/1P

Dear Mr. Grier:

This letter is forwarded to provide prompt notification of Reportable Occurrence RO-81-04/1P as required by the Millstone Unit 1 Technical Specifications.

On April 21, 1981 at 0226 during power operation at thirty-one (31) percent of rated, the main turbine was manually tripped due to excessive turbine vibration and high shaft bearing temperature. Approximately ten (10) minutes after the event, high main condenser conductivity was experienced, and the reactor was manually scrammed and isolated in accordance with emergency procedures. The Isolation Condenser System was inoperable due to post- surveillance system draining, thereby necessitating reactor cooldown through manual operation of a main steam safety/relief valve, using the torus as a heat sink. The core spray and low pressure coolant injection systems were initiated as a precautionary measure but were not injected. As a result of this means of cooldown, the reactor vessel cooldown rate was exceeded for one (1) hour.

An investigation of the cause of the event and incurred damage is in progress. A detailed description and corrective action will be provided in a 14-day follow-up report.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

E J Mroczka

E. J. Mroczka
Station Superintendent
Millstone Nuclear Power Station

EJM/HJB:ws

cc: Director, Office of Management Enforcement and Program Control, Washington, D.C. (2)
U. S. Nuclear Regulatory Commission
c/o Document Management Branch, Washington, D.C. 20555

M007
S 1/1

S
8105050529

NRC 66
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

ICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | LICENSEE CODE (2) | LICENSE NUMBER (3) | LICENSE TYPE (4) | CAT (5)

CON'T

0 1 | REPORT SOURCE (6) | 0 5 0 0 0 2 4 5 DOCKET NUMBER (7) | 0 4 2 1 8 1 EVENT DATE (8) | 0 4 2 1 8 1 REPORT DATE (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2

0 3

0 4

0 5

0 6

0 7

0 8

0 9 | SYSTEM CODE (11) | CAUSE CODE (12) | CAUSE SUBCODE (13) | COMPONENT CODE (14) | COMP. SUBCODE (15) | VALVE SUBCODE (16)

(17) LER/RO REPORT NUMBER | EVENT YEAR 8 1 | — | SEQUENTIAL REPORT NO. 0 0 4 | / | OCCURRENCE CODE 1 P | REPORT TYPE | — | REVISION NO.

ACTION TAKEN (18) | FUTURE ACTION (19) | EFFECT ON PLANT (20) | SHUTDOWN METHOD (21) | HOURS (22) | ATTACHMENT SUBMITTED (23) | NPRD-4 FORM SUB. (24) | PRIME COMP. SUPPLIER (25) | COMPONENT MANUFACTURER (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0

1 1

1 2

1 3

1 4

1 5 | FACILITY STATUS (28) | % POWER (29) | OTHER STATUS (30) | METHOD OF DISCOVERY (31) | DISCOVERY DESCRIPTION (32)

1 6 | ACTIVITY RELEASED (33) | CONTENT OF RELEASE (34) | AMOUNT OF ACTIVITY (35) | LOCATION OF RELEASE (36)

PERSONNEL EXPOSURES

1 7 | NUMBER (37) | TYPE (38) | DESCRIPTION (39)

PERSONNEL INJURIES

1 8 | NUMBER (40) | DESCRIPTION (41)

LOSS OF OR DAMAGE TO FACILITY (43)

1 9 | TYPE (42) | DESCRIPTION

PUBLICITY

2 0 | ISSUED (44) | DESCRIPTION (45) | NRC USE ONLY