EXHIBIT 6

Millstone Unit 3 Refueling Outage 6 (1999): Excerpts from Reactor Engineering Logs

Millstone Unit 3 Refueling Outage No. 6 (1999): Excerpts from Reactor Engineering Logs

(First four pages ripped out)

May 13

0000 SIGMA checks are in progress but the gripper will not pick up the dummy.
0300 **Another problem was noted with 3303C with the SIGMA interlock checks. In section 4 SIGMA is simulated over the upender and you try to lower the upender. It should not go but it does. At first we thought it may be the same problem with the gripper (i.e., SIGMA doesn't know where it is) but then we thought it could be a different problem.**
0630 **Word is that SIGMA problem may be a connector contact a wheel**
0730 **Trying to get copy of 3303C-1 Rev 4 Ch 2 from CDR – they can't find it.**
1100 **SIGMA had been believed to be OK and when doing checks, failed gripper checks.** Further checks being made
1630 SIGMA is downpowered due to electrical problem.
1843 Commenced fuel movement
1917 Overload on assembly G64 – Trip at weight of 2449
1945 SIGMA machine cannot release bundle. A SIGMA rep will be checking overload situation.
1946 2340 Per RES in SFP, definite gap observed in FA G37 (now in U-11@SFP) This is a discharge FA

May 14

615 While in containment the guys showed me a problem with the upender reservoir. It is overflowing all over the floor. It has a float valve like a toilet that sticks. We either have to fix the float valve or get permission from OPS to operate the isolation valve.
1005 Upender in SFP stuck in V position, does not go down. Movement stopped.
1024 Permission granted from SM Steve Lawhead to use bypass key for upender. Key not in containment. Obtaining key from SM and delivering to containment upender. J. Deaupre says wait on key – looking at problem.
1025 1045 Assembly H28 on SIGMA lowered down into core location R08 but not unlatched. Waiting for verdict on SFP upender.
1026 1119 44 F/As of loaded at time of upender in SFP malfunction
1155 Loss of communciations between CR and all stations
1215 Communications lost – all Ericksons system went down
1230 Cycled upender after getting bypass key – appears a torque switch was tripped due to drive chain being jogged a small amount.
1247 permission received to resume of load – upender checked out OK
1317 Refuel SRO used bypass key to get full down indication in Cont, upender FA H09
1330 SFP upender will not lower. SIGMA to [ ] to A-7 but will not latch until SFP resolved.
1500 No fuel movement in progress. 46 FAs out of core.
1950 Frame horizontal on upender – could not send to ctmt side. Pushed in on hand wheel.
2022 SIGMA machine having a problem unlatching in the upender. W going out to troubleshoot.
2033 SIGMA is going to bypass weight (take weight of) Unsuccessful. Troubleshooting other options.
2047 SIGMA has indication problems both lateral and unlatched lights on panel lit. They are going to hand crank up to 800 pounds because they believe they may be unlatched. I & C contacted to bring up tape or sleeving because it may be a repeat of a circuit problem.
2205 I & C has control of SIGMA – Standdown for 1 hour.
2300 SM concerned about rate of SFP heatup – a trend was generated.

May 15:

0480 A meeting was held at One Stop Shop on SIGMA. **SIGMA has been tested after repair and would still not work properly.** I did not attend the meeting. It's difficult to get a straight story as to what problem

is. I think the people are getting tired of being asked. One thing is for sure. They still think it's a problem with the connector.
0500 Checked the FME log on the SFP side. OK. There is a large crowd of people heading into contain ment to work on SIGMA.
1020 SFP upender needed bypass, verified FA out, received SM permission.
1115 SIGMA will not reinitialize, notified 1 stop, W, advised SM that refuel is stopped.
1200 SFP upender needed bypass key on Stop 104, verified empty, received SM permission. Slight gap observed between face 3 and 4 of F/A H84. Cart on SFP side is moving farther than should be.
1402 Step 118. Empty upender would not lower. SM granted permission to use bypass key to lower.
1402 Step 119. **SIGMA full down, received fault on panel, could not engage F/A.Raised the mast and came back down on F/A.Unsuccessful. Tried again and received [ ] grapple. SIGMA was able to come off F/A. SIGMA repairperson notified, all stop.**
1425 Update on F/A H84 – separation has been measured at 31 mils – acceptable is 40 mils max.
1440 Step 119, F/A H53 while going into upender (1 1/6ft. from bottom) lost bottom, down indication. Raised F/A and lost gripper indication. Informed SM. Able to lower and got slack cable. Asked for and received permission to get general bypass to disengage. General bypass did not work. F/A fully unlatched in upender. F/A will be put away in SFP. All work on SIGMA is stopped. Concern of work outside of procedure to unlatch. W also noticed thimble plug latch/unlatch lit. Unlatch pushed and F/A disengaged.
1510 SM halted work due to questionable containment isolation valve. (containment integrity). Steve Lozien and Dennis Barton are standing by to troubleshoot SIGMA.
1703 Fuse blew on sipping machine compressor. **Also lost SIGMA compressor**
1719 SM authorized sipping with N2 so that F/A can be lowered onto transfer machine. There is some concern that SIGMA air pressure will bleed off before the blown fuse can be replaced.
1750 SFP upender will not lower. SM authorized use of bypass key.
1900 **"Hoist slippage" error on SIGMA. Proceeding with fuel hoist. SIGMA expert does not think the problem is significant.**
1917 SFP upender will not lower. SM granted permission to bypass.
2043 SFP upender will not lower. SM granted permission to bypass
2116 SFP upender will not lower. SM granted permission to bpass interlock.
2209 SFP upender will not lower. SM granted permission to bypass interlock.
2230 Containment SIGMA crane computer showing illogical sequences of information
2240 There are 75 F/As out of the core.
2245 SFP upender will not lower. SM granted permission to bypass interlock.IAW OP3303C Precaution 3.2.2
2302 SFP upender will not lower. SM gave permission to bypass.
2334 SFP upender will not lower. SM gave permission to bypass.

May 16:

0005 SFP upender will not go down. SM gave permission to bypass.
0128 SFP upender will not go down. SM gave permission to bypass.
0130 Made a tour of SFP and Cont. . . . F/As are moving well but the SFP is the weak link. The camera inspections and the need to bypass on the upender about every other move is making SIGMA wait. Maybe the SFP is getting even with SIGMA for last night.
0142 Ass. H-38 is bowed and SIGMA having difficulty putting into upender.
0150 SM gave permission to SIGMA to use bypass. Weight and height bypassed. Ass. H-38 disengaged upender.
0204 SFP upender will not go down. SM gave permission to bypass.
0205 **SIGMA over core location J-9 nd will not give [ ] cable indication.** SM gave permission to bypass SIGMA's height and weight interlock to raise mast in an attempt to reinitialize memory. Ater raising and lowering mast, [ ] cable indication could not be established. SIGMA was moved to load test station awaiting assistance. Noticed SM, One Stop Shop and Refuel team Load. I & C and Westinghouse were contacted to investigate.
0245 SIGMA repair team arrived.
0325 **SIGMA had a problem latching the next FA also, but the experts got the thing working again.**
0350 **SFP upender will not go down. SM gave permission to bypass.**

0405 SFP crane picking up dummy to check the cable drum. Electrical maintenance noted a problem with the chain which drives the drum. This problem only occurs when operated in high speed.The chain slips. The dummy was never picked up.
0415 SM gave permission to move the FA from the upender to SFP using slow speed. There is no FA latched in containment now.
0430 SFP Upender will not go down. SM gave permission to bypass.
0450 One Stop Shop had another meeting on fuel handling problems (getting to be a nightly affair).The chain causing the problem[ ] is to the hand crank which is what caused a problem last outage.
0630 Maintenance has done a temporary fix to the chain wheel. They say we can continue moving fuel in slow speed until the temp mod to remove it is done.
0738 SFP upender could not go down to horizontal. Permission granted from shift manager to use bypass key. Upender lowered and taken out of bypass.
0739 Blanket permission to use bypass key in SFP upender to lower it from shift manager under the condition that we confirm that it contains no F.A. and that we log use of it.
0802 SFP crane will not raise off fuel assembly.
0815 On lowering F.A. H-77, brake on SIGMA not working properly. SIGMA SRO wants to wait at 1 oor of lower core plate and have maintenance look at it.
0850 SFP upender will not lower
0914 sequence deviation performed to allow placement of F/A H-77 to core location A-8
1225 Ericson communications lost approx. 1 minute
1344 Bypass key used to lower SFP upender
1410 Bypass key used to lower SFP upender
1427 Bypass key used to lower SFP upender
1429 **SIGMA getting [intermittent] indications. SRO thinks possibly water could be on air line. No impact to fuel movement.**
1446 Bypass key used to lower SFP upender
1540 SFP Upender would not raise with F/A H37
1625 Suspended fuel movement operations awaiting repair of SFP upender torque switches.
1640 SFP bridge crane tool of the hook and hung up for the duration. Preps being made to evaluate cause of upender problems.
2117 SIGMA put FA G15into upender but does not have indication that is down
2250 There is a god . . . 100 FAs out of core.

May 17

0145 We just had our nightly fuel handling meeting at the One Stop Shop. We decided to modify the spent fuel handling tool. I remembered we have a spare. Jim Beaupre was called. He says the spare is 4 feet too short (from a plant with a different SFP arrangement)
0727 SM gave permission to break communications between CR4 SFP re upender. RC will maintain coverage at SFP and communicate through normal house phones.
1315 Large cask crane hook won't go high enough
1445 Can't get tool out of water in vertical, going to use bridge crane and cask[ ] pick and work on while suspended
1657 Bypass key utilized to lower upender at SFP transfer canal Note. SM (Steve Lawehead) has given permission to the lead RE to allow bypass of SFP upender (IAW OP 3303C Step 3.2.2)without checking in with him each time. This may change when the next SM comes on. Note: Jay Ely performed a review of all our procedures as well as the SAR and verified that the alignment pin which was removed from the spent fuel handling tool is not credited anywhere.
1753 SIGNA bridge unable to get engagement light after four (4) attempts to latch onto FA H-24 at core location C-12
1905 Suspended refueling operations to allow for repair of SIGMA bridge by I & C, upender to spent fuel pool side.

May 18

0115 **SIGMA fix did not work.** All personnel are relieved from their station. I & C went and did another check of the solder joint. They are OK so it must be the connection itself. Called our nightly refueling meeting in the One Stop Shop. We decided to try to get rid o the connection by using a butt splice. If that doesn't work then the entire cable will be replaced. Estimated time to get the butt splice in is 4 hrs.
0747 Blanket authorization received from shift manager to use SFP upender bypass key to lower upender as long as it does not contain a fuel assembly. 1618 SFP upender will not raise. F/A G28 is in the upender. Shift manager gave permission to bleed the system
1635 Having dificulty placing F/A G28 in SFP location B1
1649 Upender will not raise. Contains F/A H44. SM gave permission to bleed the system.
1720 Upender will not raise
1835 recommended stand-down until troubleshooting of transfer system is complete and cause of upender problem is understood. FA G-12 is in SFPAR34. And requested or using "long pole" if necessary to manually actuate the mechanical interlock.
20ll **SIGMA needs reboot. SRO reports that they are having problems with SIGMA not lining up with core location C-14**
2016 SIGMA is going down to core position C-14
2143 Standdown recommended to allow I & C and Westinghouse to complete testing and troubleshooting of SIGMA bridge. All refueling crews standing down.
May 19 Received permission from Ray Martin to raise upender in SFP using bypass since it would not raise normally. Had run cart to full travel limit but would not raise. Bypassed interlock but frame still would not raise
0130 Upender in SFP still unable to raise
0135 Upender secured in SFP and operators sent of station
0230 Successfully raised upender in SFP. SIGMA undergoing cable replacement.
615 **SIGMA unable to go down on core location N-14**
0839 SFP upender venting system for FA D76. Significant problem this time with upender. Several attempts were necessary to raise it.
1039 CTMT upender reported that H63 bowed pretty bad.
1411 SFP RE reported that FA G24 has a slight crack on spring block mating face, definitely higher on one side. Needs further W evaluation. W evaluation determined no observable damage.
1523 Bypass key required to lower upender frame in spent fuel pool pit
1720 Will not be picking up Fuel Assembly H-04 in the core until we get someone to access the upender problems. **Getting progressively worse**
2003 SFP RE reported a black tie wrap was found on the track in the SFP transfer canal
2250 FA D79 indicated as a leaker. (Discharge asembly!)
2340{] mart sipper operator reported that signal fromnA79 indicated a small leak (500 counts). After sipping he did a purge for several minutes and then 3 blank tests for a total delay of about 15 minutes. In my turnover from swingshift I am told that the log entry from 1411 saying that FA G24 has a crack is incorrect.

May 20

616 FA D69 appears to have a damaged [ ] grid strap on face 4.The entire grid strap appears shiny so we can't tell if it is new damage or not. Face 4 was against barrel baffle. A. Ellis reviewed the tape on D69 and agreed with the above. Again recommended a close look at G55 which is the only face adjacent FA which has not been removed yet.0230 FA G58 with the source does not want to get into the core at location H15. Brought in additional lighting.

0330 requested electrical maintenance to bring additional lighting to the core. SROsays the reason for the delay in the G58 move was poor lighting.
0100 Reviewed FME log in SFP. Found one minor discrepancy.
0500 SFP RE reports SFP hoist "getting louder."
2300 OPS started GMT purge and noticed level changes in Rx cavity and in SFP. They noted that they had 1/3 turn on the gate VV but if leakage is noted after draindown may want to have engineering evaluate for additional torque on valve.

May 21

0130 Removed bypass key #50 and 59 from the SPF area. Logged the area out of the FME area, returned the keys to the control room. The keys had not been properly logged out of the control room1605 Problem with RCCA tool – were not latched at U-12 (step 122) and raised tool, which messed up the tool "sequencing." Had to hang tool and manually "reset."

May 26

1000 While working on communications gastronics sys on SF bridge, dropped wire nut into SFP
June 1 Spent fuel bridge bypass key #59 is signed out to John S. This key is to be in spent fuel RE's possession.

2025 **SIGMA needs to be re-initialed often – phantom numbers on screen and index problems.**
2230 SIGMA won't latch @ upender. They tried to raise the mast and re-initialize – did not work this time. Moved away and tried to reset – did not work.
2300 SIGMA is toes up. At present, it is latched @ upender, but will not raise or latch.

June 2

0100 On the next FA SIGMA lost light indication. Will put FA back up and try again.
0300 SIGMA quit again when trying to unlatch a FA in the core.
0315 False alarm on SIGMA, someone accidentally hit the emergency stop button.
0700 **SIGMA lost its wind again momentarily. Had to re-initialize.**
0803 **SIGMA is acting up again. Fuel movement continues.**
0815 **SIGMA blowing down air lines.**
1052 **SIGMA needed re-initialization.**
1100 **SIGMA needed rebooting over the upender. SIGMA rebooted 2d time – weird indication on screen.**
1227 **SIGMA re-initializing necessary – screen illegible & would not move (F6)**
1235 SIGMA indicates fuel down, still has 1500#. Request use of bypass to go down. Permission from SM granted.
1245 **SIGMA problems at core F6**
1308 Officially verified unlatched at F6 – coming up in bypass. **Still troubleshooting SIGMA – Re-initialized**
1555 SM gave permission to use SIGMA bypass to disengage @ RxE10. After FA is unlatched, they will raise the mast and re-initialize.
1609 Used bypass to blow out cylinders on **SIGMA – would not engage on FA in upender.**
1615 **I & C working on limit switches on SIGMA – will be approx. I hour. There is a discrepancy in position indication.**
1650 **Standdown approx. 1-2 hours.**
2145 SIGMNA had trouble unlatching. Got permission to raise mast with FA to reinitialize. Itworked.
2245 80 FAs in the core.

June 3

0310 Tried to lift FA at core locator. [ ] to get the shoehorn out. **SIGMA died in doing this.**
0400 **SIGMA is still broke. ... They are handcranking the FA off index and bypassing height & weight to try to get the FA up into the mast.**
0430 The FA is fully up in the mast. It went up on electric power. But in slow speed to avoid overload. When full up it was over a foot off on elevation.
0450 They went back to try to get the shoehorn out, but it is stuck. It did move off its initial position, rotated out, then got stuck again in a flow hole.
0515 Our plan is to place FA J51 and H50 on the bottle with a sequence deviation. Then continue loading the core away from the stuck shoehorn until a recovery plan is developed.
0550 Lost power to shufleworks connection.

0640 SIGMA is using another shoehorn now. The elbow shoehorn is stuck. They are now using the straight shoehorn. Guel movement is continuing.
1152 Refueling SRO reports a "near miss" between SIGMA and personnel directing MOV work with [ ] polar crane aux hook. SM informed.
1227 Using bypass key to reinitialize SIGMA – screen full of junk – lost brains and locked up.
1245 reinitializing SIGMA over upender.
1309 Reinitializing SIGMA over upender1407 SIGMA having difficulty with "heavy"bundle in upender.
1441 NI ch. 32 increased X 10 (14487 ct/100 sec) momentarily
1447 F/A G64 from SFP L1 is being returned to SFP Rack L1 while we try to determine what caused spike on SR 32
1530 F/A 522 is in upender, horizontal in containment. All fuel movement is now stopped! Until cause of spike and status of SR32 can be determined.
1695 Reinitialized SIGMA (didn't "find bottom")
1715 Transfer cart struck @SFP – won't traverse to CTMT, won't upend.
1750 SIGMA lost its brain (again); it's @ A-6 but is real sure that it's at H-6. Had good visual assurance that FA is lined up to A-6 – got permission to lower the FA. It worked.
2100 Lost communications, apparently due to Erickson phone network problem.
1900 Late entry – gave brief to W crew for safety standdown.
2150 Gave up on communications – stopped fuel movement.

June 4

0100 Still no communication
0200 Well the good news is that SFP RE and SIGMA are on [ ] communication with CR. Also more good news is that SIGMA and the FTs have not broken yet on midshift tonight. Bad news is that SIGMA needs more [ ] and upender operators are not hooked up yet. But we are getting close.
0435 SIGMA is having problems with their screen so they will raise FA and reinitialize.
1015 Current situation – Upender has F/A S66 in it and won't go down. SFP crane has H78 on it. H78 will be returned to M-7 in SFP.
1123 SM grants permission to use bypass key to lower upender frame in SFP.
1140 Standdown in CR, SIGMA & SRO while repairs, tests are down on SFP upender.
1552 Transient in Rakset II; suspended fuel movement while OPS assesses situation
1825 Reinitialized SIGMA, normal occurrence after 7-8 moves
2100 With SIGMA over upender and fuel assembly on hook, SIGMA lost where it was. Had to be bypassed to go to full up for reinitialization since wouldn't let operator go to their mast for initialization. Received permission from SM (Bob Smith) to bypass SIGMA.
2110 SIGMA breakers were switched off then back on again to reinitialize and find its location.2231 Sequence deviation being performed as follows: Place G14 from SIGMA into R5; move J26 from N3 to R7; "adjust" J53; Move G14 from R5 to P3; Move J26 from R7 back to N3.

June 5

0040 SIGMA reports erratic reading on their control console.[ ] Fuel movement will continue.
0150 SFP upender reports that it took several tries to get the cart to latch into position properly.
0420 SIGMA is stuck over the upender. Won't go up or down.

June 5

1000 Core reload complete.
1550 Verified correct loading. Note core location G12 is identified as having F/A H35. This is incorrect. Re-verified. F/A [ ] is H33 as per loading plan. Verified H35 in core location B4.

June 6

0015 Performed SFP videotape mapping of fuel assembly Ids. Nearly impossible to read Ids of recently discharged G assemblies. Tapes are located in RE vertical file cabinet. Found a tie wrap lying in top nozzle of fuel assembly in SFP location V41.

EXHIBIT 7

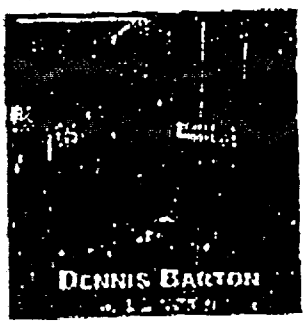
"The Daily Scorecard: Millstone Megawatts vs. Outage Barriers – All the facts, stats and at-bats for Unit 3's Refueling Outage" (May 1999)

The Daily Scorecard

Millstone Megawatts VS. Outage Barriers

All the facts, stats and at-bats for Unit 3's Refueling Outage

CONGRATULATIONS 3RFO6 MVPS!!



Triple Threat: Three talented MVPs worked many long hours on an intermittent failure of a cable o the SIGMA refueling machine in containment. I&C specialist Steve Loese, electrician Chris Ferris and Westinghouse field engineer Dennis Barton proved that perseverance overcomes technical barrier that are frustrating and challenging. The 7 conductor 53 foot-long cable was heavily contaminated and wound up on a spool at the top of the SIGMA machine, making repair efforts challenging indeed. Th cable was replaced Wednesday morning and the SIGMA machine finally managed to offload the las fuel bundle at 0902 Thursday morning. A number of other talented team members from NU ad Westinghouse participated in the job and their efforts are also much appreciated

AT-BATS
Work started in past 24 hrs.

Mode 0 Work Window
MSIV Disassembly
SWP MOV Static Test

HITS
Work completed in past 24 hrs.

Fuel Offload
Identified Leaking Fuel Assembly
Test Sample Removed from Reactor Vessel

ON DECK
Upcoming Work

Close 'A' Train Outage
Preps for 'B' Train Outage
Reactor Cavity Drain Down

STATS
Yesterday's Results

SI Exams
422 of 673 completed
Snubbers
762 of 1027 completed
Flow Accelerated Corrosion
213 of 236 completed
LLRTs
107 of 175 completed
Tank Inspections
7 of 13 completed
Check Valves
16 of 76 completed
Projects
13 of 35 completed
ALARA
Outage to Date 60.269 rem
Hours Behind Schedule
131 hrs

Nuclear Oversight Issues Stop Work Order

On Wednesday, May 19, Nuclear Oversight issued a 'Stop Work' order to Outag Management for work on all systems that could affect key safety functions, with the exceptin of work that has been verified to restore safety related equipment to the available status.

Scheduling work so that safety is maintained starts long before the outage begins Procedures OM1 (Outage Management) and OM2 (Shutdown Risk Management) describe th process by which the outage schedule is built and verified for shutdown risk. Procedure OM provides a series of action items and milestones that need to be completed well in advance b the outage, while OM2 provides a summary of the shutdown risk assessments that need to tak place for every change in key safety functions. These assessments consider the present plan conditions and any planned changes for the next 24 hours.

The following conditions initiated the 'Stop Work' order

- Some situations were identified in which work might have potentially compromised a ke safety function if it had been released as scheduled
- The long shutdown of the unit, and the shutdowns that occurred prior to the refueling ou age made the outage planning process more difficult

One of the fundamental aspects of outage management is the protection of the nuclear fuel whether it is in the reactor core or the spent fuel pool. To ensure this protection is maintained six key safety functions are continuously monitored. They are as follows

1. The ability to remove decay heat from the Reactor Coolant System (RCS)
2. The ability to remove decay heat from the spent fuel
3. The ability to add borated water (inventory) to the RCS
4. The availability of electric power sources
5. The maintenance of a level of boron to keep the reactor shutdown; and
6. Containment integrity

(continued on back)

To safely and competitively complete RFO6 with teamwork, enthusiasm, and continuous improvement.

EXHIBIT 8

Executive Summary, NNECO Nuclear Oversight Audit Report MP-3-99-A14 Refueling Activities (July 20, 1999)

NUCLEAR OVERSIGHT
AUDIT REPORT
M3-99-A14

EXECUTIVE SUMMARY

Scope

The scope of the audit was to evaluate Millstone Unit 3 Refueling Activities for Nuclear Safety, compliance with Technical Specifications, and applicable procedures. Additionally, industrial safety practices were observed.

Conclusion

Refueling personnel performance was satisfactory. Fuel assemblies were maintained in a safe condition at all times, compliance with Technical Specifications were satisfactory. Procedure use was also satisfactory. There was an adverse trend identified in the performance of the refueling equipment due to a large number of equipment malfunctions during core offload and reload. The SIGMA refueling machine, the fuel transfer system, the spent fuel building crane, and the primary communication system between the Control Room and Refueling Station all experienced malfunctions. The frequent equipment malfunctions potentially challenged the safe handling of the fuel as well as adding a significant amount of time to fuel movement.

Refueling Activities

The shift manager was always in overall control of core alterations. Permission was requested from the shift manager to commence refueling activities and use of bypasses on the Sigma refueling machine, the spent fuel crane, and the fuel transfer system. Core alterations observed were: reactor vessel head removal, upper internals removal, core offload, and core reload. The refueling Senior Reactor Operator directly supervised all core alterations. Fuel assembly movements were directed from the control room. Additionally, fuel assemblies necessarily placed in alternate core locations were tracked until correctly placed. The operations shift was kept informed of the progress of the refueling activities.

Fuel assemblies were inspected in the spent fuel building for damage and verification of the fuel assembly serial number. One damaged fuel assembly was identified. The damaged fuel assembly was a third burn assembly and was not reloaded into the core. Fuel assemblies were again inspected and serial numbers verified prior to transfer to the vessel.

Proper actions were taken when a tie wrap was noticed to have fallen into the transfer canal during work on the transfer cart. Work was stopped and the tie wrap was retrieved.

Required procedures were used for the fuel offload and reload sequence and for operation of refueling equipment. The procedures were available at all work locations.

The Sigma refueling machine experienced frequent malfunctions as did the Fuel Transfer System. The malfunctions were properly addressed by the refueling personnel.

There was one failure of the spent fuel bridge crane that had the potential to cause a fuel assembly to be suspended from the crane for a long period of time. The crane operator noticed an abnormal sound from the crane and took prompt action to place the fuel assembly in a safe condition.

The primary communication system failed on several occasions. The performance of the backup communication system, which was placed in service during core reload due to the primary system's unreliability, was marginal.

This adverse trend related to the performance of the refueling equipment was identified as an Audit Finding.

Finding

CR M3-99-2236 - "Adverse Trend in the Performance of Refueling Equipment"

During core offload and reload there were frequent problems with the SIGMA refueling machine, the fuel transfer system, the primary communication system, and one failure of the spent fuel bridge crane. These malfunctions potentially challenged the fuel's safe handling and affected the efficiency of refueling operations.

CR Owner: **Patrick Dillon, Supervisor Engineering**

Response to Audit Finding CR M3-99-2236

In response to the audit finding, Technical Support Engineering Memo MP3-TS-99-185, summarized the equipment failures, listed the apparent causes and outlined the following proposed corrective actions:

1. Evaluate potential PM program enhancements based on reviews of the following:

    a. ANSI requirements for crane inspections.
    b. Preventative Maintenance recommended by Original Equipment Manufactures.
    c. Open Automated Work Orders on fuel handling system components.
    d. CRs previously written against fuel handling system.
    e. Refuel team and Reactor Engineering logs.
    f. Historical fuel handling system corrective maintenance AWOs.
    g. New and previously-evaluated refueling equipment lessons learned.
    h. Industry Operating Experience for fuel handling equipment.

2. Visit fuel handling equipment vendors and selected plants to evaluate the design and performance capabilities of potential upgrades to the fuel handling system.

3. At least 15 months prior to RFO7, recommend upgrades for fuel handling system to management via Engineering Work Request process.

4. At least 12 months prior to RFO7, establish a schedule to complete all fuel handling system DC, PM and CM AWOs prior to core offload.

5. At least 6 months prior to RFO7, review all procedures containing pre-operational testing requirements and recommend enhancements where desired.

6. At least 3 months prior to RFO7, complete a Technical Evaluation of refueling equipment readiness.

7. Perform an effectiveness review of these corrective actions following RFO7.

The root cause evaluation was waived by the Management Review Team (MRT), based on the equipment failures being well understood by Technical Support Engineering and a formal engineering report being presented to the MRT.

Technical Specifications

Compliance with refueling technical specifications was verified to be satisfactory by the Audit Team by reviewing the surveillance procedures and verification of the performance of the surveillances at the proper frequencies .

Training

Individual Task Qualification Records were developed for each contract fuel handler prior to their working at a job position. The contractor personnel either completed the appropriate knowledge or skill section of the TQR or provided documentation of equivalency of knowledge and/or training.

Industrial Safety

Industrial safety practices were observed to be generally acceptable. There were, however, some lapses in safety practices noted by the Audit Team:

a) early in the observation period workers were noted to be stepping over the safety chain on the spent fuel bridge and were cautioned that this was not an acceptable practice, and
b) one of the refueling personnel was observed sitting on the railing of the manipulator crane and was corrected by the refueling SRO.

Deficiencies

CR M3-99-1920 - "Failure to Consistently Log Refueling Surveillance Requirements."

Technical Specification 4.9.5 requires that communication be demonstrated between the control room and the Refueling Station within one hour prior to the start of and at least once per 12 hours

during Core Alterations. The twelve (12) hour checks were performed as part of SP3672-1, however when the communications were lost or discontinued for a period of time, restoration was not always logged in the shift log.

Procedure 3303A, "Spent Fuel Bridge," states that upon completion of the Shiftly Pre-operational Checks "Request SM document that the Spent Fuel Bridge Crane is in use in the Shift Log."

CR Owner: Mike Wilson - Manager, Unit 3 Operations

CR M3-99-2235 - "Loss of Control of a Completed Surveillance"

Procedure SP3672.2, "Initial Refueling Requirements," that was completed prior to starting initial core alterations cannot be located. In addition, there is no specific written direction on how the procedure should be processed once it is completed and reviewed.

CR Owner: Mike Wilson - Manager, Unit 3 Operations

EXHIBIT 9

CR-M3-2236

6/18/99
Waiver approved by MRT on 6/16/99. Owner to provide Technical Evaluation.
Tom Gilbert

CR M3-99-2236

"Adverse Trend in the Performance of Refueling Equipment"

During an audit conducted by Nuclear Oversight, an adverse trend in the performance of the refueling equipment was identified as a Finding. The performance deficiencies were related to the SIGMA refueling machine, the fuel transfer system, the spent fuel bridge crane and the communications system. The auditors concluded that fuel assemblies were maintained in a safe condition at all times. However, the CR proposes that a root cause evaluation be performed to determine if any programmatic issues exist that could result in equipment failures and potentially challenge the safe handling of fuel.

Technical Support Engineering is aware of the equipment malfunctions that occurred during RFO6 and suggests that a root cause investigation to identify potential programmatic issues is not needed because of the following reasons:

1. The unreliability of the SIGMA control console was well known prior to RFO6. The existing console is an antiquated computer that has caused problems in the past. Many other plants have upgraded their control consoles and Unit 3 had previously submitted an EWR to replace the console during Cycle 7.
2. One of the major contributors to the SIGMA breakdowns was a connector in the cable between the control console and the mast. This cable was replaced and the connector was eliminated during the core offloaded window. The connector was needed because Westinghouse delivered the wrong length cable during a previous modification of the mast. The cable and connector appeared to be acceptable during RFO5.
3. The manual chain drive for the spent fuel bridge hoist was removed by a temp. mod. during the core offload. This feature had been designed by Westinghouse and installed prior to RFO4. An EWR was initiated during Cycle 6 to replace the chain drive mechanism, but the parts were not available prior to RFO6. Maintenance Services adjusted the chain drive mechanism immediately prior to core offload in an effort to ensure its reliability. Unfortunately, the poor design of the mechanism resulted in failure. This mechanism had also failed in RFO5, but the System Engineer initially recommended reinstalling the mechanism to determine if the failure in RFO5 was due to poor installation technique. The new design eliminates the chain and is scheduled to be installed in Cycle 7.
4. The fuel transfer cart holddown latch springs were jamming at the end-of-travel position in the fuel pool, preventing the latch from opening completely. These springs were replaced with a different design during the core offloaded window. Subsequent operation of the springs was satisfactory. However, Maintenance also discovered that the cart was rubbing on the tracks for approximately 6 inches prior to the end-of-travel. Health Physics and Engineering are already planning to pull the cart from the canal during Cycle 7 and repair the problem. Additionally, the latch does not return to center when the cart is leaving the fuel pool. This problem will be more thoroughly investigated when the cart is removed.
5. The communications system failures resulted from insufficient coordination with Purchasing in ordering the equipment desired by Reactor Engineering. The equipment supplied did not meet the needs of Reactor Engineering and the Ericsson phones were used as a last resort.
6. The fuel handling equipment preventive maintenance AWOs were all performed in accordance with vendor manual instructions. Additionally, a PaR engineer and the system engineer performed a walkdown of the fuel transfer system prior to core offload and no deficiencies were found. The transfer cart was also transferred to containment with the canal drained and no deficiencies were noted.

In summary, the company management and virtually every plant department realize the need to handle nuclear fuel safely and efficiently. Many plant departments worked together for 5 months prior to RFO6 to perform the PMs specified by the fuel handling equipment OEM and also performed the necessary troubleshooting and repairs when deficiencies were found. Management supported design changes, where justified, to ensure that the fuel could be handled safely and efficiently. Maintaining the equipment is always a major evolution for the Maintenance and Health Physics departments and is frequently given

lower priority than work required to keep the plant on line. In spite of this, work was prioritized adequately and all PM AWOs were completed prior to the start of core offload. Upgrading the equipment to resolve performance problems is usually expensive and also requires significant time and effort by many departments. The need to upgrade some of the equipment and improve the preventive maintenance program has been reinforced by the poor performance of this equipment in RFO6. However, it is unlikely that a time-consuming root cause investigation will find any unknown programmatic deficiencies that contributed to these performance problems.

Signature on file	10/21/98	10/30/98	98-60
Form Approved by	Approval Date	Effective Date	SORC Mtg. No.

AR No. 99009441	**CR Form** Initiation	CR No: CR M3-99-2236

Section 1: To be completed by initiator (please type or print)

Organization identifying condition: Nuclear Oversight	Discovery date: 6/9/99 Discovery time: 0900	Affected Unit(s): 1☐ 2☐ 3☒ C☐	System #:

1. Condition description (including how condition was discovered, organization creating condition, what activity was in progress when event was discovered):

Adverse trend in performance of the refueling equipment.

During core off load and core reload there were frequent equipment problems with the SIGMA refueling machine, the fuel transfer cart system, the primary communication system, and one failure of the spent fuel bridge crane. These malfunctions affected the efficiency of the refueling operations and potentially challenged the safe handling of the fuel. Had the equipment failed in a manner such that a fuel assembly could have been damaged or been unable to be moved to a safe location, severe challenges to nuclear fuel safety could have occurred.

This is an Audit Finding, a response to Nuclear Oversight is required within 30 days.

Continuation Sheet ☐

Component ID.:
Method of Discovery: Nuc. Oversight (RP 4, Att. 1)
Source Document:

2. Immediate corrective action taken
none required

TR# AWO# Continuation Sheet ☐

3. Recommended corrective action
Perform a root cause analysis of the equipment malfunctions to determine potential underlying programmatic cause(s).

Continuation Sheet ☐

4. Initiator Requests Follow-up: ☒ Y ☐ N
Initiator Name: David Andersen Time: 0900 Phone No.: 3155
Initiator's Signature: [signature] Date: 6/9/99 Cost Control Center 84F

Engineering Disposition: Y ☐ N ☒ Name/Dept. of Dispositioning Engineer: ① ② Name/Dept.
Requested

Supervisor Name: Donald Gorence Time: 10:20
Supervisor Signature: [signature] Date: 6/9/99 Phone No: 5529

Section 2: To be completed by Operability/Reportability Screening Designee

1. Does CR have an actual or potential effect on plant or personnel safety, operability, reportability, reactivity management or plant operation?

If continuation sheets (RP 4-1, Page 7) are required, identify the section being continued by section number.

CR Form
Initiation

CR No: M3 99-2236

Section 2: To be completed by Operability/Reportability Screening Designee

1. (continued)

☐ Yes or Don't Know
☒ No

If yes, describe reason:

(1)

Designee: [signature] Date: 6/9/99 Time: 1035

If continuation sheets (RP 4-1, Page 7) are required, identify the section being continued by section number.

Condition Report

Section 3: To Be Completed By Shift Manager

CR No: **M3-99-2236**

1. Personnel Safety
   - ☐ Does not affect personnel safety
   - ☐ Actions taken to protect personnel
2. Operability assessment (Describe basis in comments)
   - ☐ Condition does *not* affect SSC operability
   - ☐ Condition made SSC inoperable but operability restored
   - ☐ Condition makes SSC inoperable
   - ☐ SSC not currently required to be operable but condition must be corrected prior to Mode ____
   - ☐ With the existing condition reasonable expectation of Continued Operability exists, Operability Determination initiated (RP5)
3. Reportable?
   - ☐ Yes; per: _____
   - ☐ No
   - ☐ Reportability Determination Required
4. Reactivity Management
   - ☐ Yes; Notify Reactor Engineering
   - ☐ No
5. Comments Including any immediate corrective actions taken):

Shift Manager: Time: Date:

Section 4: MDMRIT Screening Sig Level: [X] 1 [ ] 2 [ ] 3 — Risk Significance: 3

1. CR Title: Audit Finding: Adverse trend in performance of the refueling equipment

2. CR Owner: 3MGRTCHSUP Inv Due Date: 7/2/99

Comments:
Level

- ☐ MDMRT closed to immediate corrective actions
- ☐ CR closed to TR/AWO# _____, no further documentation required
- ☐ CR closed to CR# _____, no further documentation required

CA Department: **Linda Precopio** (Signature) Date: **June 11, 1999**

If continuation sheets (RP4-1, Page 7) are required, identify the section being continued by section number.

Condition Report

Section 5: Evaluation Summary - CR Owner

CR No: M3-99-2236

1. Event Summary (For Level 1 CRs attach the Root Cause Analysis; For Level 2 CRs include organization(s) responsible for the condition, what happened, activity and process being performed, why did it happen.)

a. Organization (s) Responsible:
Technical Support Engineering is responsible for assuring that fuel handling equipment is ready to perform its function. The responsibilities include establishing the preventive maintenance program requirements and recommending equipment modifications to assure the system will handle fuel safely and efficiently.

b. What Happened:
The fuel handling system was not reliable during RFO6. There were varied and numerous equipment problems that occurred which indicated that the process of preparing the fuel handling system for refueling was inadequate. Nuclear Oversight classified this adverse trend in the performance of the refueling equipment as an audit finding.

c. Activity and Process Being Performed:
This condition was identified during fuel handling operations in support of RFO6.

d. Why did it Happen (Apparent Cause):
See attached memorandum MP3-TS-99-185.

Continuation sheet ☒

2. Similar Situations or Generic Implications
Does the condition apply to other NU units, other trains, or for other situations?
☐ Yes, describe applicability and recommended actions.
☒ No, explain.
This CR applies to the Unit 3 refueling equipment. The Unit 2 refueling equipment operated reliably during the core onload.

Continuation sheet ☐

3. Recommended actions not accepted and why
MRT determined that a root cause analysis of the equipment malfunctions to determine potential underlying programmatic cause(s) was unnecessary.

Continuation sheet ☐

If continuation sheets (RP 4-1, Page 7) are required, identify the section being continued by section number.

Condition Report

Section 5: Evaluation Summary - CR Owner (Continued) | **CR No: M3-99-2236**

4. Action Plan

CA#: 1	Description of Action/Effectiveness Review	Tracking No: 9900994/1-02

Evaluate potential PM program enhancements based on reviews of the following: a) ANSI requirements for crane inspections, b) PMs recommended by OEMs, c) open AWOs on components, d) CRs against system, e) refuel team and RE logs, f) historical CM AWOs, g) refueling lessons-learned, h) industry OE.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS (cc 11/22/99) MGRCOMSYS (1)

Manager Accepting Action	Alert Group: ~~3MGRTCHSUP~~ (835C)	Assign. Type: CACA	Due Date: 2/29/00	
	Name: VINCE SPAMER	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 2	Description of Action/Effectiveness Review	Tracking No: 9900994/1-03

Visit vendors and other plants to evaluate design and performance of potential refuel equipment upgrades.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS (cc 11/22/99) MGRCOMSYS (1)

Manager Accepting Action	Alert Group: 3MGRTCHSUP (835C)	Assign. Type: CACA	Due Date: 11/30/99	
	Name: V. SPAMER	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 3	Description of Action/Effectiveness Review	Tracking No: 9900994/1-04

Recommend upgrades for fuel handling system to management via EWR process.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS (cc 11/22/99) MGRCOMSYS (1)

Manager Accepting Action	Alert Group: 3MGRTCHSUP (835C)	Assign. Type: CACA	Due Date: 12/15/99	
	Name: V. SPAMER	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 4	Description of Action/Effectiveness Review	Tracking No: 9900994/1-05

Establish a schedule to perform all PM, CM and DC AWOs prior to RFO7.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS (cc 11/22/99) MGRCOMSYS (1)

Manager Accepting Action	Alert Group: ~~3MGRTCHSUP~~ (835C)	Assign. Type: CACA	Due Date: 4/1/00	
	Name: V. SPAMER	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

Assignment Type Coding: (Investigation (CATI), Remedial (CACR), Compensatory (CACC), Corrective (CACA), Corrective to Prevent Recurrence (CACP), Effectiveness Review (CATE), Other (CATT)

If continuation sheets (RP4-1, Page 7) are required, identify the section being continued by section number

Condition Report

Section 5: Evaluation Summary - CR Owner (Continued) | **CR No: M3-99-2236**

4. Action Plan

CA#: 5 | **Description of Action/Effectiveness Review** | **Tracking No:** 9900944-06

Review all fuel handling procedures containing preoperational testing requirements and recommend enhancements, where desired.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS @ 7/26/99 MGRCOMSYS (1)

Manager Accepting Action	Alert Group: 3MGRTCHSUP (835C)	Assign. Type: CACA	Due Date: 9/30/00	
	Name: V. Spruill	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 6 | **Description of Action/Effectiveness Review** | **Tracking No:** 9900944-07

Complete a Technical Evaluation of refueling equipment readiness.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS @ 7/26/99 MGRCOMSYS CATE 9/15/99 (1)

Manager Accepting Action	Alert Group: 3MGRTCHSUP (835C)	Assign. Type: ~~CACP~~	Due Date: 12/15/00	
	Name: V. SPRUILL	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 7 | **Description of Action/Effectiveness Review** | **Tracking No:** 9900944-08

Perform an effectiveness review of this corrective action plan.

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS @ 7/26/99 MGRCOMSYS (1)

Manager Accepting Action	Alert Group: 3MGRTCHSUP (835C)	Assign. Type: CATE	Due Date: 8/31/01	
	Name: V. Spruill	Sched. Ref: N/A	Mode: N/A	
	Signature:	Officer Signature		

CA#: 8 | **Description of Action/Effectiveness Review** | **Tracking No:** 9900944-09

Ensure implementation of required corrective actions to prevent recurrence by modifying corrective action plan as necessary to support RFO 7 fuel handling activities

AITTS SYSTEM/PROGRAM INDICATOR 3303/FHS (1)

Manager Accepting Action	Alert Group: MGRUNITSYS	Assign. Type: CACP	Due Date: 12/12/99	
	Name: G. Swidor	Sched. Ref: RFO7	Mode: N/A	
	Signature:	Officer Signature		

Assignment Type Coding: (Investigation (CATI), Remedial (CACR), Compensatory (CACC), Corrective (CACA), Corrective to Prevent Recurrence (CACP), Effectiveness Review (CATE), Other (CATT)

If continuation sheets (RP4-1, Page 7) are required, identify the section being continued by section number

Condition Report

Section 5: Evaluation Summary - CR Owner | **CR No: M3-99-2236**

5. Investigation Completion Certification:

Initiator requested feedback	☒ Yes	☐ No	
Initiator advised of proposed resolution	☒ Yes	☐ No	☐ NA
Initiator agrees with proposed resolution	☒ Yes	☐ No	☐ NA

Investigator: J. F. Beaupre/ x4823 (Name/Phone) Signature: _____ Date: 6/24/99

6. Corrective Action Plan Approval:

a. Level 1, 2, and 3 Condition Reports:

CR Owner or designee

(Name): V. SPUNAR Signature: _____ Date: 6/24/99

b. Level 1 Condition Reports:

Responsible Director

(Name): P. Grossman Signature: _____ FOR PBG Date: 6/24/99

Section 6: CR Package Completeness Checklist - Corrective Action Coordinator

Root Cause Evaluation	☐ YES	☒ N/A	
Other Required Documentation	☒ YES	☐ N/A	
Trending Information	☒ YES	☐ N/A	

Corrective Action Coordinator (sign): _____ Date: 6-28-99

If continuation sheets (RP 4-1, Page 7) are required, identify the section being continued by section number.

Condition Report

Section 7: Documentation of Additional Reviews (Level 1 and 2 CRs only) CA Manager	CR No: M3-99-2236	
Multi-disciplined management review required?	☐ NO	☒ YES
Meeting Date: 7/21/99 ☒ Accepted	☐ Accepted with comments	
☐ PORC or ☐ SORC review required?	☒ NO	☐ YES
Meeting No: _____ Meeting Date: _____ ☐ Accepted	☐ Accepted with comments	

1. Copy of Level 1 Risk Level 1 or 2 CR sent to NSAB Staff? _____ Yes
Initial

Comments:
☐ MRT recommends placing on Nuclear Network

Section 8: Implementation Completeness Review (CAD Staff) (1)

Closure documentation received for CAP completion
_____
INITIAL

CR Owner Approval Assignment Complete (1)
_____
Date

Unit Corrective Action Department: _____ _____
Signature Date

CR status changed to "CLOSED"? (1)
_____
Initial

If continuation sheets (RP4-1, Page 7) are required, identify the section being continued by section number

Attachment 10
Condition Report Evaluation Checklist
(Sheet 1 of 1)

This checklist should be used by the Corrective Action Coordinator when submitting a CR action plan to the Corrective Action Department.

CR # M3-99-2236	Corrective Action Coordinator

() indicates section of RP 4-1

	Area	Yes	N/A
1	All pages in CR package have CR number on them.	✓	
2	Event Summary (5.1) contains (1) What occurred, (2) Organization(s) creating condition, (3) Activity and process being performed, which created the condition and (4) Why it happened. (Level 1 may refer to Root Cause, N/A for Level 3)	✓	
3	Generic Issues (5.2) are identified and acted on.		✓
4	For action recommendations not accepted a legitimate reason is provided. (5.3)		✓
5	Corrective Actions stand on their own, are clear, and can be implemented by the assigned owner.	✓	
6	Corrective Actions properly filled out. No omissions of Assignment Type Code, Owner, Alert Group, signature, due dates, Sched ref code, or mode. (5.4)	✓	
7	For Level 1 CRs the following assignments are included: CATPR, compensatory actions if CAPTR not complete, and Effectiveness Review. (5.4)	✓	
8	Adequate documentation included to support completed actions. (5.4)		✓
9	Initiator feedback provided, if requested. (5.5)	✓	
10	Investigator signature. (5.5)	✓	
11	CR Owner signature. (5.6)	✓	
12	Responsible Director Signature (Level 1 CRs only) (5.6)	✓	
13	Required documents in package and Completeness checklist filled out. (Root Cause, LER, Reportability/Operability/MRFF Determinations with package if applicable). (6)	✓	
14	Trending Information complete. (6)	✓	
15	Corrective Action Coordinator Signature. (6)	✓	

Comments

EXHIBIT 10

Transcript, Deposition of Michael C. Jensen (May 11, 2000)

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	:	Docket No. 50-423-LA-3
Northeast Nuclear Energy Company	:	
Millstone Nuclear Power Station, Unit No. 3	:	MAY 11, 2000

DEPOSITION OF MICHAEL C. JENSEN

CERTIFIED COPY

Kathryn Orofino
Shea & Driscoll, LLC
Court Reporting Associates
16 Seabreeze Drive
Waterford, Connecticut 06385

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	:	Docket No. 50-423-LA-3
Northeast Nuclear Energy Company	:	
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16 Seabreeze Drive
Waterford, Connecticut 06385

A P P E A R A N C E S :

NANCY BURTON, ESQ.
147 Cross Highway
Redding Ridge, Connecticut 06876

For Connecticut Coalition Against Millstone
Long Island Coalition Against Millstone
The Intervenors

WINSTON & STRAWN
1400 L Street, N.W.
Washington, D.C. 20005-3502
BY: DAVID A. REPKA, ESQ. and
DONALD P. FERRARO, ESQ.

For Northeast Nuclear Energy Company

NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555
BY: Ann P. Hodgdon, NRC Staff Counsel

ALSO PRESENT:

Dr. Anthony C. Attard
David W. Dodson
Laurence T. Kopp, Ph.D.
David Lochbaum
Victor Nerses
Gordon Thompson, Ph.D.

INDEX OF EXAMINATION

INDEX OF EXHIBITS

(None offered at this deposition)

* * * * *

Deposition of MICHAEL C. JENSEN, a witness in the above-entitled action, taken at the request of the Intervenors pursuant to 10 CFR Section 2.740a before Kathryn Orofino, a Notary Public within and for the State of Connecticut, at the Mystic-Noank Library, 40 Library Street, Mystic, Connecticut, commencing at 1:40 p.m.

* * * * *

STIPULATIONS

The deposition is to be used for discovery or as evidence in this proceeding only; objections or motions to strike will not be considered to be waived except as to matters of form; the Deponent will be given a right to read and sign the transcript when it is complete; the original of the transcript will be forwarded to the deposing attorney who will provide the opportunity for the witness to read and sign; and the original will be filed with the Commission in accordance with the Commission's rule of 10 CFR part 2.

M I C H A E L C . J E N S E N ,

of Northeast Nuclear Energy, P.O. Box 128, Bldg. 475/2, Waterford, Connecticut, 06385-0128, a nonparty witness in the above-entitled action, having been duly sworn by Kathryn Orofino, a Notary Public within and for the State of Connecticut, was examined and testified on his oath as follows:

* * * * *

MS. BURTON: Do you want to state the stipulations so we can be consistent.

MR. REPKA: Sure. This is a deposition of Mr. Jensen that's being conducted by the Coalition Against Millstone. It's to be used for discovery purposes and possible evidence in this proceeding only. The witness should be given an opportunity to read and sign the transcript when it's prepared. Objections or motions to strike related to the testimony here today will not be considered to be waived.

And with that, we're ready to begin.

MS. BURTON: Okay. Good afternoon, Mr. Jensen.

THE WITNESS: Good afternoon.

EXAMINATION BY MS. BURTON

Q Can you tell us what role you have been assigned to in the matter of the pending application to

reracking of the Unit 3 spent fuel pool.

A The reracking in the Unit 3 spent fuel pool is headed by a project team. They perform all of the necessary calculations and engineering and paperwork associated with that.

My group, reactor engineering group, provides a review function for the spent fuel project group. So the bottom line answer is we provide review functions.

Q Okay. And what about you; what is your role?

A I'm the supervisor and I supply the staff to perform those reviews.

Q So would it be fair to say that you are the -- you lead this reactor engineering group which is analyzing and submitting and following through with this application?

A I don't know that "analyze" is the correct characterization. We review any analysis that may be provided with the documentation.

Q Did you assist in the preparation of the amendment application?

A No.

Q At what point did you first become involved in the amendment process?

A We're involved in it in an engineering aspect, not in the application aspect. The application

is performed by another group. The project group leads it. I'm not sure if they do it themselves or not. We reviewed conceptuals and the engineering diagrams, the construction diagrams and things like that.

Q And when did you begin your work on this particular amendment?

A It would have started approximately 9 to 12 months ago.

Q Who else is on your team?

A Well, I have a staff of seven. I have five -- actually, four engineers, a person who has the title of analysis, but he works in the plant thermodynamic response area, not in this area, and I have two technicians.

Q Would you like to give me their names?

A Okay. The technicians are Kathy Emmons and Sheila Stark. The engineers are Kent Wietharn, Jeffery Camp, Bob Berchert, Steve Claffey. And the analyst is John Gibson.

Q Thank you.

The license application itself has a reference to ANSI N210-1976.

A If you say so.

Q I believe it does.

Now, I wonder if you know if -- if you're

aware that that section has been replaced in the intervening time by another section?

A No, I'm not aware.

Q So you would not know necessarily -- well, I guess that presumes that you haven't analyzed the materials pursuant to the new section of the ANSI code?

A No, because as I said, we don't analyze. My group does not analyze. We review the proposal in an engineering sense and in a use sense. We end up being the major user of the new racks that are going in, so the type of review that we would conduct is does it meet our needs. We wouldn't review it for -- I'm assuming you're alluding to the quality of materials or things like that.

Q Not the quality, of the standard that may be --

A No, we don't review it against that.

Q So you're assuming that the change would meet all the standards. The only question for you is would it suit the need for the plant?

A Yes.

Q I see.

And I assume you have an opinion as to whether or not the application as submitted does suit the need of Millstone?

A Yes, the application -- yes, that is my opinion.

Q What is your opinion?

A That it meets the need of the plant as submitted.

Q Does Millstone Unit 3 have present capacity for a full core off-load in its own spent fuel pool?

A Millstone 3 currently does have the capacity. The storage racks that are there, there are 756 available locations, which I believe 496 currently are occupied. The core holds 193 assemblies.

Q Would you happen to know how the NRC staff came to its determination that the plant lacked full core off-load capacity as of the time of its issuance of a finding of no significant impact last year?

A No, I don't know how they would come to that. Currently we can offload the whole core. We have the capacity to do that.

Q Now, you have mentioned that you work -- that you work with -- it's the reactor engineering group?

A I am the supervisor of the reactor engineering.

Q I'm sorry. Supervisor of --

A Reactor engineering.

Q Okay. I got that wrong.

When was that group formed?

A We had a reorganization approximately a year ago. And prior to that, each unit had its own reactor engineering group. In the reorganization of the engineering department, it was determined that reactor engineering would become a site group. Unit 1 was no longer in need of that type of engineering service, and Unit 2 and Unit 3 both being PWR's and closely related, it was determined that a site group would be a more efficient and effective way to organize.

Q So prior to your present assignment, what was your previous position with Millstone?

A I was previously the reactor engineering supervisor of Millstone Unit 3.

Q And in that capacity, you became familiar with the events at the spent fuel pool at Unit 3?

A My tenure there was a short one. It lasted probably five months prior to the reorganization in July of last year. I was there from February of 1998.

Q Now, you have been asked, apparently, to participate in this discovery process?

A Yes.

Q And, in fact, you have participated by providing certain information in the form of an affidavit and also materials, references to materials

and documents?

A I or my staff have, yes.

Q And, in fact, you have identified particular participation in Interrogatories E-1, E-4 and F-1 that the two Intervenors filed, correct?

A I believe that to be true, yeah.

Q I wanted to ask you particularly about Interrogatory F-1.

A Okay.

Q Do you have a copy of that?

A I don't remember them by number. Yes.

Q Now, this is one of the ones that you indicated that you provided information for in the submission; is that correct?

A Yes.

Q And this is the interrogatory that asks for identification of all instances of errors at Millstone or other nuclear plants in managing, moving, placing or tracking fresh or spent fuel and all pertinent documents thereto; is that correct?

A That's true.

Q Could you please tell us what process you followed to gather the information that you used to respond to this request.

A I assigned Kathy Emmons, who is a reactor

engineering technician, to determine which documents would, in fact, meet the request, and she provided the documents.

Q And can you please tell us what instructions you gave her in terms of collecting the information that would be responsive to that request.

A It was as simple as I stated it; please determine the documents that meet this request. There are several tools available to her to do this search, and she can seek help from organizations such as licensing and the plant operation staff.

Q I think you identified her as a technician previously a few minutes ago, but then you ascribed a different title to her?

A No, she is a reactor engineering technician.

Q Okay. And what are her ordinary responsibilities apart from this special assignment?

A A reactor engineering technician is a person typically who takes care of some of the administrative requirements of the group, they normally take care of SNM accountables. They are the SNM bookkeepers.

Q What is SNM?

A Special Nuclear Materials.

They also, during refueling outage, play very active roles in the refueling of the particular unit.

Q And how long has -- and could you spell her name please, Kathy.

A Emmons.

Q Emmons?

A E-M-M-O-N-S.

Q How long has she been at Millstone?

A I couldn't say with any accuracy, but it's in the neighborhood of six or seven years.

Q Do you know what her qualifications are professionally?

A I -- I can find out precisely. I know she has a bachelor's degree and a master's degree, I believe it's in -- the master's degree is in safety. She has 23 years of experience, all of it with Northeast Utilities, the bulk of that being with Connecticut Yankee, where she was an operations technician, and she was a reactor engineering technician for Connecticut Yankee prior to coming over to Millstone.

Q And that was six or seven years ago?

A Yes, it was.

Q Now, there is a description here of 11 events in response to Interrogatory F-1?

A Yes.

Q And who compiled this list?

A I believe the attorneys compiled it.
Q From what information?
A From the information supplied by Kathy Emmons and others.
Q Who are the others?
A I don't know.
Q Did you provide any of the information?
A Directly, no.
Q Did you attempt to retrieve any of the information in response to this interrogatory?
A What do you mean by "retrieve"?
Q Go into some kind of a record repository --
A No.
Q -- database.
A No, that was Kathy's job. That was her assignment. I did review the list.
Q Now, do you know where she obtained -- where she was able to locate these documents?
A I do not know the exact method that she used to search out these documents, no.
Q What is your best understanding of where she went to retrieve these documents?
A Well, there's several databases that she could interrogate. There is a program called LIST, which is Licensing -- I forget what the I stands for --

Search Tool.

Q That's internal at Millstone?

A Yes, it is.

Q And what would that encompass?

A That encompasses correspondences to the NRC, LER's, anything referencing new regs. or reg. guides, things like that. It's a historical database, it's not a database that's kept current in today's time frame. It's typically six months to a year behind chronologically.

Other databases she could search could be the Corrective Action database.

Q Where is that kept?

A That's also within the Northeast Utilities' LAN System.

Q Land?

A Local Area Network. It's a computer. You know, in terms of computer, you ask where it's kept, I know it's kind of -- it's on a computer hard drive someplace within the LAN system.

Q And I'm sorry, it's called the Corrective Action --

A Yeah --

Q -- database, did you say?

A It's a Corrective Action database. We used

to call them ACR's, Adverse Condition Reports, and now they are called Condition Reports, and it's a database that documents all of those.

Q And I assume the LIST is also a computer system?

A The database is a computer database.

Q And what other resources?

A There are hard copy sources. I don't know which ones currently exist or in what state. They are typically kept by departments for historical reasons. Before LER's, we had Plant Incident Reports. Licensing normally would track and trend those things.

Q Now, when you say "licensing," do you mean the licensing department?

A Yes.

Q And what would their tracking system be called?

A That would be a better question for Dave Dodson than me. I don't know the methods that they would employ, whether it be hard copy or a computer based system. I know they want to go to a computer based system. I don't know that it is right now.

Q What else exists in terms of the database that's responsive -- in terms of what's responsive to

this question?

A I can't think anything else, although that doesn't preclude her from using something I haven't said.

Q Now, do you know if she went into each of these databases to collect the information?

A No, I did not have a checklist and I did not go down something like this with her specifically, but it's within her skill to know that those databases exist. She would have queried them.

Q But you didn't specifically ask her, for instance, if she went to the historical records and hard copy?

A No, I did not specifically ask her that.

Q Now, can you tell me in what form the information was presented -- I gather it was presented to you, you accepted it, and then sent it along to the attorneys?

A Essentially, yes.

Q What form was it presented to you by her?

A It would be in a list of information that she found, and I would take a look at the list, do these items, in fact, meet the -- I guess you're calling it an interrogatory, but it's a request for information. Does it meet the request? And I reviewed that as yes,

it meets the request, and then forwarded it to the attorney.

Q So, in other words, it was a list, it wasn't a collection of the documents themselves?

A It was a collection of documents, but there was a cover sheet. "Here's the documents contained herein" would be the type of list that sat on top of it, and I reviewed that list.

Q Now, is that the same list that appears here in response to Interrogatory F-1?

A It was a shorter list.

Q Okay. How was it that it was shorter than this list?

A I -- I'm not certain which ones we did not supply but that someone else may have supplied.

Q Well, I understood from your affidavit, Mr. Jensen, that you are the individual responsible for responding to this interrogatory?

A Yes.

Q But yet information was provided to fulfill this request and you don't know who provided it or where it came from?

A That's true. However, I did review the response to this interrogatory and I did review this list, and this list is germane to that question or that

request for information.

Q Were there any items that you deleted from any of the sources that came to you responding to this request?

A None.

Q Sitting here today, you can't be sure that this list is complete, can you?

A No. I don't know that anybody could.

Q Well, what would be required -- what process would be required to be followed to determine the complete and full answer to this interrogatory?

A Well, again, I don't know that you can have the absolute, but as I said, all the databases known to us to be queried.

Q Are you familiar with the requirements, the standards, the thresholds for recordkeeping at Millstone with respect to information that would be responsive to Interrogatory F-1?

A I guess I don't understand your question. What --

Q Well, the fact that there are 11 titles indicated here suggests that somebody made a determination that these were reportable events in some sense, they were reported and recorded, there is a record of them.

A Uh-huh.

Q So I'm asking you to tell me if you're familiar with what the requirements are, what the criteria are to the event to be recorded so that they enter any of these various databases that you just identified?

A I'm somewhat familiar with the criteria for these things to enter the different databases, yes.

Q And could you tell us what the criteria are?

A Well, the Corrective Action database, basically in the ACR, as they are formally known, or CR, Corrective Action, that's filled out are entered into the database. There is no filter or no exclusion from that database.

The LIST database is a compilation --

Q Excuse me, I didn't mean to interrupt, but to go back to corrective actions --

A Yes.

Q -- these corrective actions are internal to Northeast Utilities, correct?

A Yes.

Q They are not automatically and necessarily reported to the NRC?

A The NRC has access to them, but they are not, if you could say, overtly given to them. They have

access to them. It's a database they can review or search on or anything else.

Q And what is the requirement? Is it internal or is it a federal regulation that there be a keeping of these corrective actions materials?

A I don't know what the requirements are to keep records on corrective actions or CR's. There is a requirement to have a corrective action program.

Q Okay. I interrupted you, but could you continue.

A Okay. The other database is LIST. I've remembered what the "I" was. Licensing Information Search Tool. That is a compilation of all known correspondence to the NRC, which would -- the Licensing Event Reports would be a subset of, but if we have any correspondence with the NRC on issues, that it is incorporated into this database.

Q How long has that database been in existence?

A If my memory serves me right, it was created in the early '90's. It was a project that was contracted out.

Q And was there something else that performed a similar function prior to the early '90's?

A Not a similar function. This particular piece of software and database were put together for

ease of search. Prior to that, hard copy was the only way we maintained records as far -- and again, Dave Dodson could give you more information from the licensing standpoint.

Q How far back does the Corrective Action database go?

A From the inception of the Corrective Action Program, which would be mid 1990's.

Q Prior to that, there were Adverse Condition Reports?

A Right. Same program, just a different title for the report.

Q And when did the station begin to commence keeping --

A Mid to early '90's.

Q Same thing for adverse conditions?

A Right. They are the same thing. We just -- the only change in the title was we wanted to encourage people to use this system, so the word "adverse," people felt, well, it's really not that bad, maybe I shouldn't write anything on it. We wanted to take that potential barrier to reporting things away to encourage people to write all conditions that they felt needed management attention.

Q But prior to beginning to keep the data in

the Corrective Action database or the Adverse database, where was the same information kept?

A That type of -- well, actually, I'm not sure. When someone had a problem, they went to their supervisor, they tried to correct it through a normal organizational type of effort. There was no documentation, or at least a program or formal documentation that I know of.

Q So is it possible that there were events that today would be reported under the Corrective Action program that would not -- that may not have been reported earlier?

A The possibility exists, yeah.

Q But there might be no records in any of the databases of some events that may have occurred that would otherwise be reported to these databases that now exist?

A I would have to say that that possibility exists, because in today's environment, we encourage the reporting of the slightest concern, so we have a tremendous database being built. And it's basically a live on-line database that's kept current within a few days. Prior to that, there was no such mechanism.

Q And you say "prior to that." Could you establish a date?

A Again, that's the mid to early '90s that the Corrective Action program was --

Q That would have been '92, '93, '94?

A Somewhere around in there.

Q I wonder if you happen to have with you the various reports that correlate with the list that is responsive to Interrogatory F-1?

A I personally don't, but I'm sure that --

MR. REPKA: Are you referring to the documents listed in the April 20th response?

MS. BURTON: April 4.

MR. REPKA: Okay. April 4 lists the event.

MS. BURTON: Lists the event.

MR. REPKA: Right.

MS. BURTON: And then I have --

MR. REPKA: And then April 20th --

MS. BURTON: -- the production of master lists.

MR. REPKA: All right. We're with you.

MS. BURTON: So what seems to be is 38 through 47.

MR. REPKA: Could be.

BY MS. BURTON:

Q Is that correct?

A Yeah, if you're asking me if I have copies of those with me, I do not.

Q But you are familiar with the actual reports?

A I'm not familiar with detail, I'm familiar with the actual report, the general description of the report.

Q And I would assume that would be the case, especially if your name appeared on one of them?

A I might have more detail if my name appears on one of them.

Q Okay. Well, I'd like to take a moment to go through some of these --

A Sure.

Q -- beginning with Number 38, as appears on the Licensee's Document Production Master List as Attachment A responding to our Request for Production.

A Okay.

Q And Number 38 is titled "Millstone 1 Adverse Condition Report M1-97-0082. A radiated fuel assembly stored in damaged fuel container in control rod storage rack January 14, 1997."

A Yes.

Q Now, according to this report, apparently at Millstone 1 an irradiated -- do you have it before you, Mr. Jensen?

A Yes, I do.

Q Okay. So you can see that the description is that an irradiated fuel assembly MS-508 is stored in a damaged fuel container in a control rod storage rack?

A Yes.

Q And that a comprehensive assessment of the acceptability of this storage configuration and location may not have been performed?

A Yes.

Q And that this question was raised during inspection of a spent fuel pool.

And dropping below here to Item 5, it seems to indicate here that MS-508 was dropped and damaged in 1974?

A Yes.

Q Since that time, it has been stored in a damaged fuel container?

A That is correct.

Q So in other words, that condition remained between 1974 and 1997; approximately 23 years?

A Yes.

Q Now, if you could look at Paragraph 11 on the front page of that document.

A Yes.

Q It says, "How discovered performance of

RE-1071."

A Yes.

Q Do you know what "RE-1071" means?

A I'd have to look it up. I can tell you the activity that was being performed. The --

Q But you can't tell me what "RE-1071" means?

A No.

Q Below that number 12, there's a question on this form, "Does ACR have an actual or potential adverse effect on safety, operability, reportability or plant operation." Do you see that?

A Yes.

Q And there's a check mark here under "Yes"?

A Yes.

Q Now, the individual who signed this report, can you identify that signature?

A Yes. Daniel J. Meekhoff, M-e-e-k-h-o-f-f.

Q Now, would it be fair to say that it was the determination of that gentleman that this phenomenon involved a safety, operability, reportability, or plant operation?

A What that indicates is that he has answered the question that's asked exactly the way it's worded there; "Does this ACR have an actual or potential adverse effect on safety, operability, reportability or

plant operations." He checked yes.

Q Now, can you please tell us what the standards and criteria are with reference to that particular question on this form, which is the Adverse Condition Report Form.

A All Adverse Condition Reports at that particular time were brought to the on-shift manager for an initial review that -- those particular people are trained in Code of Federal Regulations on what's reportable, what's not. They also have NRC operator licenses, so they understand plant operations to a high level of detail.

They also know whether the -- with those two particular credentials, they also know whether the particular piece of equipment is operable or not. And whether it affects safety is both an issue of personal safety, equipment safety and nuclear safety. And they are also trained on that.

Q So would it be fair to conclude from the information shown on here under Section 12 that this would be a reportable event to the NRC since it's checked "Yes" to that question?

A No. Because that's checked "Yes" does not mean it's reportable. Any one of those items -- safety operability, reportability, or plant operations --

could result in a yes, so it's not fair to assume that anything checked "Yes" is reportable.

Q Do you know if this particular event was reported to the NRC?

A It was not reported in the form of a License Event Report, it was reported to the resident inspector. They were notified of this when we had performed the fuel pool inspection.

Q Now, you say "we." What was your role in this particular event?

A Mike Biterelli (ph) really was the initiator of this, and I was his supervisor at the time.

Q When you say initiator, what do you mean by initiator?

A He's the one that wrote up the report. He's the one that wrote up this ACR.

Q And at the time you were his supervisor?

A I was his supervisor.

Q Now, at Page 2 of this report under Section 4 it says, "Is the ACR" -- that means Adverse Condition Report, I assume?

A Yes.

Q -- "reportable"?

And it's checked off here, "Uncertain." Do you see that?

A Uh-huh.

Q And the determination as to whether it was reportable at that time would have been made by the gentleman who signed here, the same, or is that a different gentleman?

A This --

Q Daniel Meekhoff, I guess the same as before?

A Yes. Once the person signs on Item 12, page 1, that says yes, there could be an actual or potential, that same person goes through this checklist on page 2, or the following page, and goes through line by line to check to see that the plant conditions are noted at the time in case they are relevant in determining whether it is reportable or not or as to whether it affects safety or not.

And they also review the plant conditions and the actions taken once the discovery is made to make sure they are sufficient for the current time. And then he goes through the rest of the list, and "Reportable" is part of this checklist.

Q Do I recall you saying that there was no License Event Report filed technically with regard to this incident?

A I'm unaware of one.

Q But you're saying the NRC was notified

somewhat less formally?

A The resident was notified of our finding, yes.

Q Do you know if the resident notified superiors of the NRC?

A I don't know.

Q Do you recall the name of the resident?

A Not off the top of my head, but I could determine it if you need it.

Q Now, at page 3 of this same document, Section 2-B, what is the ACR significance level? What is checked here?

Are we looking at the same page? Oh, 4, I'm sorry. The pages were sticking. 2-B.

A Yes.

Q What is the ACR significance level?

A Originally?

Q It could be A, B, C or D, right?

A That's correct. Originally it appears to be checked C, and that appears to be stricken, initialed, and B is now checked.

Q Now, do you know when that revision was made?

A No, it's not dated.

Q And what are the different levels of significance in terms of seriousness?

A Yes, A being the most serious, D being the least serious. Each requires a different action or different level of action.

Q Do you know why it was revised from C to B?

A I believe it was with discussions with the management that it required a little more attention. After I had checked records, I could not find whether that particular fuel assembly had been assessed in the condition which we found it.

Q And why was it important to have that information?

A It's important to have that information because you're concerned about all the components in the spent fuel pool, that they are, in fact, in a safe condition, and I could not locate the documents that clearly stated that the condition in which we found this damaged fuel assembly in the damaged fuel container as an acceptable condition.

Q And what did you do as a result of the determination that you couldn't find that information?

A We did an investigation as to, actually, the events that took place that resulted in the damage to the fuel assembly, how it arrived in the condition it was in the container, and then we determined that we should do an analysis on that particular condition

relative to its ability or its K-effective status.

Q Now, you're talking about the damage going back to 1974?

A Yes.

Q So you looked for all the records of that event --

A Yes.

Q -- in 1974?

And what did you find?

A No records at all.

Q What records should have been there?

A Well, we were looking for some sort of documentation concerning the recovery of that fuel assembly, and we couldn't find any.

Q Do you have any idea why you couldn't find any?

A No. Either they weren't generated, or if they were generated, they weren't kept, they weren't kept as a hard copy in the operations' file or the engineer's file, nor in the nuclear document services.

Q Do you know what the circumstances were that led to this Adverse Condition Report being filed 23 years later, or the discovery of the -- or rediscovery of the condition?

A Through my investigation, I know how the fuel

assembly ended up in the condition it was, yes. And it was my group that was doing a fuel pool survey that identified this as a potential adverse condition.

Q And when was that?

A The survey? The survey -- this was in the middle of the survey, so the date of this ACR would be in the middle of a two-week process, so it would be January of 1997.

Q And what was the reason that such a survey was undertaken at that time?

A We were doing a video survey of the spent fuel pool for a couple of reasons. I had just become the reactor engineering supervisor of Millstone Unit 1 at that particular time, and there were questions about the spent fuel pool configuration control.

The special nuclear material within the spent fuel pool was, in fact, inventoried and highly accountable. The remaining things that were in the pool, we have some spent instruments and there were some end fittings of some control blades that we had processed earlier in the pool.

So in order to completely reconcile the inventory of the pool and to check on the cleanliness status of the pool, I had a video inventory done of the whole pool, both of the top of the racks and down under

the racks.

Q Now, this was after the decision was made to decommission Unit 1?

A No. We had entered a refueling in 19 -- in late 1995, and in mid 1996, I -- I took over the -- or was it '95. In mid 1996, I took over the reactor engineering department.

Now, this was during -- the plant was shut down in order to create our response to NRC-5054-F letter requesting that we supply information that would prove that we are in compliance with the requirements to operate the plant; our technical specifications, the safety analysis report and any NRC commitment.

Q Jumping ahead a couple of pages, if you could, in that document to where it says at the top, "Reportability Assessment."

A Yes.

Q It says that this fuel assembly was damaged when it was dropped onto the SFP floor in 1974?

A That's correct.

Q It was subsequently recovered into the failed fuel container 18 months later?

A Yes.

Q I wonder how that was determined if there were no records from that time.

A Yes. We called up the engineer who was in charge of the recovery. His name is Paul Merry. We located him down in Florida and we interviewed him and obtained this information.

Q Did you ask him, or was he asked if he had provided written records of that event and where those records might be?

A He said he had no records of that.

Q He had no records, or he did not make records?

A He said he had no records. We did not ask if he made any. We assumed he didn't make any if he didn't have any.

Q Why would he have any if he wasn't working at the plant?

A He was working at the plant at this time.

Q I see. You mean he didn't have records at the plant? He had been working at the plant continuously --

A Yes.

Q -- from 1974 at least until '97?

A No, he was not involved in the -- if you will, rediscovery of this condition. He had left the company probably six or seven years prior to that.

Q Right. So when he was questioned about this,

he was no longer working for the company?

A That's correct.

Q So why would he have the documents with him?

A Sometimes people retain personal documents.

Q This would not be a personal document, would it, the records of this dropped fuel assembly?

A Whether it's a personal document or a company document would be the choice of the person who develops it, I suppose. We asked him if he was in possession of anything related to this, and he said he was not.

Q So you're saying that individuals who work with the spent fuel pools at Millstone have an option of writing reports of events and keeping them as personal records, not having them maintained at the station? Is that what you're saying?

A No, you're not fairly characterizing it. I'm saying some people have copies of records that they consider personal copies of records. And we were asking him if he had anything in his possession relative to this event, and he said he did not.

Q In the third paragraph on that same page is a reference to efforts to be made to measure to determine the effect of a cavity drain down event.

A Yes.

Q Do you know what that refers to?

A Yes. There are several things in the particular configuration that we found that were of concern to us and we wanted to evaluate their significance.

In this particular situation, the fuel bundle was not fully seated in the canister because -- I'm going to have to go into a lengthy technical description of how we put it in the container, if you want.

Q Well, I'm really more interested in the cavity drain down event.

A Well, okay, assuming that you're accepting that it's not fully seated in the fuel canister, it, in fact, sits approximately 8 to 10 inches above a normally fully seated fuel assembly in a storage rack, so it sits a little higher than a normal fuel bundle.

Now, in a drain down event such as a cavity seal failure during refueling or something like that, the cavity can, in fact, drain to a point. And that point is known. The point is above fuel that is fully seated in the fuel racks.

We wanted to ensure that water was still covering this fuel assembly for two reasons; to ensure that there was adequate heat removal, which was a minor concern because of the age of the fuel assembly, and

the more important was that there was adequate amount of shielding not to significantly change the estimated radiation doses for a drain down, which we determined that there was.

Q And also you determined that this condition ultimately was not reportable?

A I believe that to be the case, yes.

Q And by that it means not reportable to the NRC?

A Yes, under Title 10 of the Code.

Q And the next page, CR Action Closeout, there's a check box for significance level with three options; Level 1, Level 2, Level 3.

A Where is this?

Q This would be the page at the top of which it says "CR Action Closeout."

A Yes. Let me look at something. Yes.

Q Significance Level 1, 2 or 3?

A Yes.

Q And which one is checked?

A 1.

Q And is that the most serious?

A Yes.

Q And whose determination was it that this was a Level 1 significance event?

A That was mine.

Q And can you explain why?

A Yes. During the time intervening between filling out this form and the actual creation of this ACR, we changed the forms and we changed the categorizations of the ACR's from an A, B, C, D level to a 1, 2, 3 level. Remember this was originally checked as C, upgraded to a B, and then this particular system changed its categorizations.

So when we went to close it out, the most appropriate significance level of the new process was a Level 1.

Q So, in other words, on one page of this document Level 1 is checked as the most significant; another document shows there were four options. It was first checked as C, and then B. But what you're saying now is that the correct and accurate one would be the highest level, whether it was three options or four?

A That's correct.

Q And what standards and criteria did you apply when you made the determination that this was a Level 1 in terms of significance?

A Within RP-4, both the version that categorizes Levels A, B, C, D, and I believe it's Revision 4 that went to a 1, 2, 3 scaling of

significance, there are descriptions within the procedure that aids you in determining the significance.

Q What is the RP-4?

A Pardon?

Q What is the RP-4?

A RP-4 is a procedure designation. "RP" stands for "Reports," and this is the fourth procedure in the reports chapter of the administrative procedures.

Q Now, is that internal at Millstone or is that NRC imposed?

A This is -- that procedure is internal to Millstone to come into compliance with the requirements for a Corrective Action program.

Q Can you explain to me why, if you found this to be of Level 1 significance, it was not also found to be reportable to the NRC?

A Not all Level 1 significant CR's are reportable to the NRC.

Q Well, what was it about this that led you to make the assessment that this was not reportable?

A It didn't meet the criteria within Title 10 of the Code.

Q What criterion?

A That would be 10 CFR 50.73 and 74.

Q Okay, but translating that to this particular situation, what was it missing? It was not a safety issue?

A No, it wasn't, because the investigation led to understanding how the condition got to where it was, and all the elements that were of concern to us, the potential radiation impact, the cooling of the particular damaged fuel assembly, the reactivity of the damaged fuel assembly, were all assessed. And we did not meet any of the thresholds to cause this to become reportable.

Q Now, is this particular assembly in the same location today?

A Yes.

Q And it's still elevated --

A Yes.

Q -- above others?

A Yes.

Q Is it still elevated at the position that's shown at Attachment 6?

A Where in this attachment are you referring?

Q Attachment 6 at the bottom, "Because MS-508 is stored in a damaged fuel container, its elevation is approximately 11 inches higher than the elevation for a fuel assembly that is fully seated in a fuel storage

rack."

A Yes.

Q Now, there are different documents that are referenced, I believe, in this report, but they are not included. Do you know where those materials are; various assessments, for instance, of General Electric? Attachment 6 references a GE analysis, I believe.

A Memorandums from Millstone can be had in the correspondence files, and anything to do with technical specifications, the FSAR, IE Bulletins, and GESTAR can be found in Nuclear Document Services.

Q If we were to make a specific request for these documents, you would probably be able to find them, or somebody would?

A Yes.

Q Thanks.

Let's look at Number 39, which is entitled "Adverse Condition Report M1-96-0646. Spent fuel assembly not fully seated in suspense storage rack," et cetera.

A What was the date on that one?

MR. FERRARO: This is October 7, 1996.

A What is the ACR number?

BY MS. BURTON:

Q This is what it looks like.

A Okay, yes.

Q If you could please turn to the third page of that where it says under "Safety Function. Fuel assembly MSB-062 is not fully seated in its storage rack. This condition is documented in APR MP1-96-0646. An inspection of the spent fuel pool was performed on October 10, 1996, to identify any similar conditions. During this inspection 56 assemblies that are not properly seated were identified."

Do you see that reference?

A Yes.

Q "The cause for improper seating is in Boraflex racks. 12 bundles elevated due to channel fastener engagement and four bundles elevated by channel button engagement with debris possible in one location. In boron carbide racks, 37 bundles elevated due to channel fastener engagement, and three bundles elevated due to channel button engagement."

Do you have any personal familiarity with this particular report?

A Yes.

Q And what can you tell us about that?

A Again, this inspection was performed by my group and, again, it was a video inspection. These particular bundles we found at first, the first bundle,

as you cited, was not fully seated in the storage rack, which prompts the question, are there any others like that.

Upon review, we found several assemblies that were not fully seated. In BWR fuel, each fuel is channeled, which is different than PWR fuel. In order to appropriately seat the fuel within the core, there is channel fasteners upon which there are springs, so when you bring four fuel assemblies together, the springs space the four fuel assemblies apart.

They are outside the normal dimensional width of the fuel assembly. In other words, they are on the outside of the channel. When placing these -- apparently, when placing these in the fuel storage racks, these channel fasteners cause an obstruction, and when the fuel assembly was set down, the fuel channel's fasteners supported the fuel assembly, and they were approximately four inches higher than a fully seated fuel assembly.

Q Now, do you know when they were installed?

A We went back and reviewed the records to see if there were any commonalities between these fuel assemblies, and we did not find any gross commonalities between these fuel assemblies. We did find that the majority of these fuel assemblies were placed in their

current locations by one NNECO employee, or by the last refuel contract vendor.

Q When, please?

A They were -- different bundles were placed at different times.

Q What is the range of time?

A The range of time would be over the last six to eight years.

Q The last six to eight years before 1996?

A Yes. The vast majority of them did occur within the last two years prior to 1996.

Q But not necessarily all at the same time?

A No, not at -- no, not all at the same time.

Q Certainly not all at the same time?

A Positive that they were not placed all at the same time.

Q And you're certain, because you have all the records that would document when and --

A Yes.

Q -- how they were placed?

A As part of our special nuclear material inventory control, any movement of a fuel bundle is documented.

Q However, there's an exception that we just went through which goes back to 23 years?

A What exception?

Q Well, there was -- may have been documentation, but you couldn't find it?

A Oh, we have documentation of that fuel assembly. I mean, we didn't lose track of it. What we don't have documentation of is how it was broke and recovered.

Q I see.

And what do the records indicate as far as why these particular assemblies were placed the way they were?

A There's nothing in the documents that alludes to the fact that they were not fully seated. I mean, it -- the records we maintain is on their location. And they are in their documented locations.

Q Now, why was an assessment of fuel assembly dropped from six inches performed in this case?

A The -- as I had said, the fuel channel fastener exists on the outside of the channel and it is holding the bundle up by interfering with the rack itself. Should a seismic event occur, there is nothing that would guarantee the fuel bundle would remain the approximately four inches above its fully seated position, so it did have a potential during a seismic event to drop that distance.

Q Okay. Let's look at Number 40.

A In which document is that?

Q That one is entitled "License Event Report."

A April 19th.

Q "Movement of new fuel assemblies over the spent fuel pool resulted in a condition outside of the design basis of the plant."

MR. FERRARO: If you give us the date, it's easier.

MS. BURTON: April 19, 1996.

Q It looks like this.

A Yes.

Q Do you have personal familiarity with this one?

A No.

Q This was also Millstone Unit 1?

A Yes. It predates my taking over the group by approximately four to five months.

Q Now, apparently from this report on March 6, 1996, "With the plant shut down and the reactor was in the cold shut-down condition, it was determined that new fuel assemblies had been carried over irradiated fuel assemblies in the Millstone Unit 1 spent fuel pool."

"These fuel assemblies were lifted over the

spent fuel pool following receipt and inspection of new fuel assemblies during operating cycle 15, as they were transported with the reactor building overhead crane from the fuel inspection stand to the fuel preparation machine in the spent fuel pool."

A Yes.

Q Now, it says further here, "Moving new fuel assemblies with the reactor building overhead crane introduced the potential for the new fuel assembly to be dropped in a height of approximately 28 feet above the top of the storage rack. This has resulted in a condition outside the design basis of the plant and is reportable pursuant to 10 CFR 50.73A to 2B."

It also says, "This event was not promptly reported since the event is historical in nature and the condition does not currently exist."

Can you explain what is meant by that, that the event is historical in nature and therefore was not promptly reported?

A I can only give you my understanding of the situation, since I wasn't involved in it, nor was I involved in the follow-up to it.

When we receive new fuel for cycle 15, the fuel is brought up to the refuel floor, placed in an inspection stand. An inspection is done and a channel

fastener is placed over the fuel assembly. The fuel assembly is then taken with the overhead crane over to a new fuel elevator in which it is lowered into the pool.

It is my understanding that the fuel assembly was brought over the spent fuel pool from the inspection stand to the new fuel elevator, which creates a drop height of 28 feet.

Q And this is a condition outside of design basis?

A The drop analysis at the time was for a drop of a fuel assembly that was being held by the refuel machine, which means it's already in the fuel pool, so, yes, it -- it appears to be a condition outside of our design analysis.

Q Well, when actually did it occur; do you know?

A The fuel, I believe, was received in late September and early October of 1995.

Q But it was not reported at that time?

A I believe that to be the case, yeah, by this document.

Q Although at that time, it was a reportable event?

A Yes, anything outside your design base is

reportable.

Q Can you tell us why it was not reported at the time?

A No, I don't have any information on that.

Q Was Millstone ever penalized for not reporting this event in accordance with the standards for License Event Report?

A I don't know what the NRC deemed with this particular LER, whether it was -- whether they followed up a NOV or a fine, I'm not aware.

MR. REPKA: I don't think it's established that it wasn't reported, that there was a noncompliance with the reporting requirements.

BY MS. BURTON:

Q What is the reporting requirement, Mr. Jensen, for a condition outside the design basis? How soon does that need to be reported, how soon is that required to be reported?

A I would have to look up in the Code of Federal Regulations 50.73 to take a look at the words to tell you where the thresholds and the dividing lines are.

However, a historical event that currently does not exist is less important to the NRC than a condition that currently exists. So since this was

claimed to be historical in nature and did not currently exist, the -- the reporting requirements are less than if it currently existed. But we can look that up, if you like, in the Code.

Q Okay. Page 3 there's a statement here, "Cause of Event. The cause of this event is personnel error in the failure to define a load path for the transport of new fuel."

A Yes.

Q Was that information reported to the NRC when the License Event Report was eventually reported?

A I'd have to take a look at the LER to be specific, but I would see no reason to omit that.

Q Let's look at Number 41, which has a date of November 17, 1995, Adverse Condition Report ACR-06385, "Fuel assembly placed in MNP-1 fuel pool in wrong orientation." Do you have that, Mr. Jensen?

A 06385?

Q Yes.

A Yes, I do.

Q Now, this was not reported to the NRC according to Item 4 on the second page of that sheet?

A Yes, that block is checked "No."

Q So it was not reported?

A As far as I know, it was not reported.

Q Now, page 3 has a description of impure water clarity. Do you see that reference? Under "Action Description," it says in part, "fuel pool filter" --

A "/Demin was placed in service" --

Q "/Demin," D-e-m-i-n.

A -- "to improve water clarity."

Q And then it says, "Poor water clarity contributed to this event"?

A No, it does not. It says "to improve water clarity."

Q Below that, doesn't it say "poor water clarity contributed to this event"?

A Yes, it does.

Q And on the next page under Section 7 --

A Yes.

Q -- there's a handwritten notation here, is there not, "Improved water clarity makes verification of bundle orientation easier to perform"?

A Yes.

Q And that would have been noted by Mr. P.R. Blomberg, whose name appears at the bottom?

A Yes. Well, I don't know that he wrote that. I mean, his name exists at the bottom. Paul Blomberg was, at the time, an event analyst when he was with the company.

Q I wonder if you could please turn to this page.

A Yes.

Q This appears to be a report by a J. Nemin --

A Nemin, but yes.

Q -- Nemin, who according to this report, spotted the misorientation.

A Yes.

Q And apparently in this case, a fuel bundle was supposed to be oriented to the southwest, but was loaded to the southeast. It was then withdrawn and reoriented?

A Yes.

Q And apparently in this case there was an issue as to the clarity of the water?

A Yes.

Q And there's -- there are several observations here. The first one includes the statement, "The next time I was on the bridge, I noticed that the surface of the water in the reactor cavity and FFP was constantly rippling. This made it more difficult for all but the mast operator to see through the water. The mast operator was using water box attached to the mask."

A Where exactly are you reading?

Q That's Observation 1, and it goes on to

Observation 2. "The water in the SFP was murky. There appeared to be a lot of" -- and then the word is C-R-U-D in capital letters, "suspended in the water. This made it more difficult to see through the water in the SFP. Clarity of the water improved over the next few days."

And it goes on to say under Observation 3, "The SFP underwater lighting is uneven and not as good as the reactor cavity."

Do you know Mr. Nemin?

A Yes.

Q Have you discussed his observations with him?

A No. Again, this particular CR predates me.

Q Well, apparently, according to his report, the combination of rippling water surface, murky water and lighting made it hard to see the clamp, which if it had been noted in time, could have been brought to the attention of the operator so that the orientation would have been installed correctly.

Do you know what conditions existed that caused this apparent murkiness in the water?

A No.

Q Do you know if the lighting was changed after this report was filed by Mr. Nemin --

A Yes, it was.

Q -- on November 24th, 1995?

A Yes, it was. The lighting in the Millstone Unit 1 spent fuel pool are lights that are hung from the curb, and they can be positioned -- depending upon what area in the pool you are working in, you can bring more lights over to that particular area if you need them.

Q Was it ever determined what caused the murkiness in the water?

A I don't know.

Q Was anything done to the water to clear it?

A That I don't know. I don't know if it naturally became clear, or whether a filtering unit or the installed spent fuel pool purification system was used.

Q Now, would that be something that would be within the jurisdiction of the chemistry department at Millstone?

A The chemistry department could make those recommendations, the reactor engineering could make those recommendations. The operations department would be the department that would implement them.

Q Do you know who was the head of chemistry at Millstone at that point in time, November 9th, 19 --

A If my memory serves, I believe it was

Dave Wilkins.

Q -- '95?

Dave Wilkins. Who is the present head of chemistry at Millstone?

A Bob Griffen is the manager for the site.

Q So in terms of the chemistry department addressing an issue of murky water, if that were to happen today, that would be under his jurisdiction ultimately?

A If the chemistry department addressed it, yes.

Q Let's now go, please, to Number 42 dated October 4th, 1985, "Millstone Unit 2, Plant Incident Report. Fuel assembly lowered onto fuel assembly in spent fuel pool."

A I'm going to have to look at that other index again.

Q Yes.

Now, this apparently involves an incident at Unit 2 where there was a safety implication involving potential damage to fuel assemblies, correct?

A That's what it says, yes.

Q Now, according to this report, this was an incident not reportable to the NRC?

A Apparently who evaluated it checked "Not

Reportable."

Q And checking "Not Reportable," does that end the path of reportability?

A This is back in 1985. We had Plant Incident Report forms. And I'm not sure whether that ended it or not. That particular process has been replaced for many, many years.

Q Now, what apparently happened in this case was that the spent fuel pool platform crane operator unloaded the weight of a fuel assembly onto another fuel assembly?

A That appears to be the case, yes.

Q And the error is attributed to personnel error?

A It says operating error, yes, as a cause of failure.

Q And it says here under Corrective Action, "Placed A-040 into location B31 and instructed operations and RE personnel performing fuel movement to pay closer attention when placing fuel in SFP storage racks"?

A Yes.

Q Now, apparently the fuel assembly that was being lowered weighed the equivalent of 1,135 pounds -- excuse me -- the weight of 1,405, the wet weight

equivalent?

A Are you reading that from something?

Q I'm reading that from this page.

A Okay, yeah.

Q Would it be your understanding that there was a potential safety aspect to this event?

A There is the potential for one, yes, but I believe, as I read this -- again, this predates me also -- fuel handling and SNM procedures were reviewed and no procedural inadequacies were identified. Assembly A017 was reviewed for mechanical integrity and no problems identified.

Q So in this case, really, there was no corrective action that was deemed to be appropriate to be implemented?

A Other than the corrective action stated.

Q Number 43, Adverse Condition Report ACR-0710, "Spent fuel pool crane operator went to wrong location. Stopped by checker. April 27, 1995."

A Yes.

Q Are you personally familiar with this?

A No.

Q Page 3, it says that no LER was required to be filed with the NRC?

A The "No" box is checked. Yes, that is

correct.

Q So would it be fair to assume that this was not reported to the NRC?

A Not in an LER fashion. However, as I stated before, the resident inspector is typically informed, but I cannot confirm he was in this case, but in most cases similar to this, they are told.

Q And they could be told informally in person without there being any documentation?

A Yes, that could have been.

Q But you don't have any personal knowledge?

A This also predates me.

Q We have just a couple more to go through here.

The next one is Number 44, Millstone Unit 3 Plant Information Report 394-079, Fuel Misplacement, April 27, 1994.

A Yes.

Q Do you have that, Mr. Jensen?

A Yes, I do.

Q And it says, "Here is a description of the event. Fuel assembly moved to wrong location and momentarily placed on another fuel assembly. Description of suspected cause if known, human error."

A Yes, that's what it says.

Q Now, where it says under 2, Safety Implications, somebody has written "NA." Would that stand for not applicable?

A That's typically what NA stands for, yes.

Q Below that, under "Event Category," it's checked, "Not reportable to NRC"?

A That's correct.

Q If you would turn to the second page, it says here under 4, "What could be done or changed to prevent this problem from happening again." And there are four notations here; "Rig an underwater light from breech crane to illuminate those racks; 2, continue to check MTF" -- is it BS map?

A Versus -- yes, that's a material transfer form versus the map.

Q " -- prior to lowering fuel assembly; 3, minimize conversations on the bridge; 4, dual verification of fuel movement."

Now, under 5, "Any other information you consider important. I have allowed myself to get overextended with too many projects. Blackness testing, perhaps, BTRS resurrection mode," and what is that next?

A "Mode zero alternate cooling."

Q "Also I've been up since 0130. I came in to

work 0500." Do you know whose signature appears under that statement?

A I do not recognize it. However, I would assume it's Butch Bornt, who printed his name at the top.

Q Okay. And this is dated April 27, 1994?

A Yes.

Q Can you tell us what blackness testing is?

A Blackness testing is a method used to determine absorption ability of a neutron absorbing material. In industry perspective, it's a test done on Boraflex to measure the neutron absorber, the Boraflex.

Q Now, on the third page of this document in the description of the event, apparently Mr. Bornt is an engineer?

A I don't know Butch Bornt.

Q He's listed here as an engineer.

A I see that.

Q Now, there's a statement, "We had completed move 48 on MTF Number 3-94-005 F/AB 39 from cell AA-30 to Y-41. I was holding a conversation with Tom concerning mode zero alternate fuel pool cooling. I forgot to cross out the cell we had just loaded."

And then it goes on, "I mistakenly told the PEO to go to cell Y-41 and forgot to cross check the

MTF and the map. We moved over cell Y-41 and I visually checked to verify that the cell was empty. However, due to the poor lighting in that area, I did not see the fuel assembly. The PEO also checked, but he, apparently, did not see it either."

I'm sorry, but what is the PEO?

A Plant Equipment Operator.

Q "The PEO lowered the fuel assembly and the hoist stopped. We raised the fuel assembly, moved it away, and visually inspected the cell again. I also double checked the MTF and the map and then discovered my error. The time was approximately 0850."

It goes on to say, "I now realized that we should have halted fuel movement and notified the shift supervisor when the misplacement occurred, and that the following corrective actions were taken. I reviewed STAR principles and reminded myself that this activity is a prime candidate, repetitive, monotonous," et cetera.

Can you tell us what the STAR events of those are?

A It's a philosophy or a way of doing business that was implemented in the mid 1990s to preclude human errors. And STAR is an acronym that stands for Stop, Think, Act and Review. It's a method by which you can

enhance, correct deliberate actions.

Q And are the people who work in the spent fuel pool -- do they go through any programs at Millstone that acquaint them with those principles and seek to assist them in their work responsibilities?

A These principles are taught to everybody at Millstone. It's a -- it's an expectation from management that these principles be used.

Q Is it a particular issue in the spent fuel pool where there are repetitive and monotonous activities?

A It's a good principle to use in any physical activity, so yes, it's a good principle to use in the spent fuel pool.

Q Now, if you could turn to this page of that document.

A Yes. I've got a couple of them that look like that. What's it say at the bottom? 2. Okay. I got it.

Q There's a question, "What could be done or changed to prevent this problem from happening again?" And the response is, "Provide lighting from under the spent fuel pool bridge in order to be able to see if there is an assembly in any location in the pool. The only lights available are on the pool walls, and the

location I was going to was in the corner of the fuel rack furthest from the wall."

And then it goes on to say any other important information -- I'm sorry -- "Any other information you consider important." And the information has been provided here, "The engineer should have a better way of keeping track of the fuel assemblies." And I would gather that a J. Cote, C-O-T-E prepared this --

A Yes, Jeffery.

Q -- this report April 27, 1994.

Do you know Mr. Cote?

A I know who he is. I do not know him.

Q And the next page after that is a -- this is a questionnaire that asks for other pertinent information where it says, "No Stop Work Order given or notification to supervisor to lighting was poor in this rack section. Some confusion may be created by the number of procedures in use." And what does it say after that?

A "For plant in 1 ACP."

Q What does that mean?

A For plant procedures and 1 Administrative Control Procedure.

Q Now, does that have reference to the activity

of the fuel movement that's the subject of this particular document?

A Yes.

Q Do you know what those procedures would be referring to?

A I can only assume that they involve the operation of the equipment and the building itself to set it up for moving. And the Administrative Control Procedure would be the Special Nuclear Material Accountability Procedures.

Q Now, that statement came from an investigator?

A It appears to, yes.

Q And do you recognize that signature?

A No, I don't. And I don't see any other name on that piece of paper.

Q Possibly Jack Dart?

A Jack or Dale.

Q But that name wouldn't --

A No.

Q -- be known to you?

Let's look at Number 45. License Event Report 87-019-00, Misoriented fuel assembly, July 8, 1987." Do you have that, Mr. Jensen?

A Yes, I have that.

Q Do you have personal familiarity with this?

A No.

Q Now, it says, "Description of the event on June 12, 1987, at 1915 hours. While unloading the reactor core during a scheduled refueling outage, a fuel assembly was found to be 90 degrees out of the proper orientation. After notification of appropriate management personnel, the fuel assembly was moved to the spent fuel pool and core unloading continued."

It goes on to say, "This event is reportable per 10 CFR 50.73A 22."

It goes on to say, "Cause of Event. During core loading operations in the 1985 refueling outage, LY2729 was not loaded in the proper orientation. Following core loading, the reactor core was verified per RE 1077 reactor core verification. This procedure involves videotaping the reactor core, verification by reactor engineering and quality assurance personnel that the, quote, 'as loaded,' unquote, core is identical to the core map supplied by the General Electric Company, and reconstruction of the core from the videotapes by an independent third party from the quality assurance organization, incorrect orientation of LY2729 was not identified during performance of this procedure."

Would you have any insight as to why it was not identified during performance of the procedure?

A No, I do not have any information as to that.

Q This is Number 46. "Millstone 2 Plant Incident Report, fuel handling incident, March 18, 1985."

A Yes, I have that.

Q Do you have that, Mr. Jensen?

"Description of Event. While handling fuel in refuel pool lowered assembly G-21 on top of assembly age 16 which was in the north upender (ph). Apparently, this was deemed not reportable to the NRC?

A That block is checked.

Q And let's now look at Number 47.

MR. REPKA: 47. You're right. 47.

MS. BURTON: "Abnormal Occurrence Report. Inadvertent drop of an unchanneled fuel assembly, September 27, 1974."

MR. REPKA: Do you have a copy we can glance at? It doesn't look like we have a copy in front of us.

MS. BURTON: Yes. Thank you.

Q Now, this event involves the inadvertent drop of an unchanneled fuel assembly from the main fuel gravel to the floor of the spent fuel pool, correct?

A Yes.

Q And I would assume, given the date, you didn't have personal familiarity with this?

A No, I didn't. However, it is the one we investigated. That is the fuel bundle that is in the damaged fuel canister.

Q Oh, I see. This is related to the very first one?

A Yes, it is. That's the LER when the fuel assembly was initially damaged.

Q I see. And in this case, as a precautionary measure, plant management ordered an evacuation of the entire reactor building?

A That's done by procedure on all events of this nature.

Q And why is that?

A The -- because you cannot determine the significance of the damage at the time the incident occurs. We don't want people to sit there and try to determine the damage.

Q In other words, there is considered to be significant risk of damage -- risk of significant damage if there is a requirement of complete evacuation of the entire reactor building?

A It's precautionary because you don't know

what the damage is. If you were to fail the cladding, there can be a release of gas, and there is no need for someone to be in that environment. In situations like this, there's really nothing that can be done as an immediate response. If damage has occurred, you cannot repair the damage from the refuel floor, so as a precautionary measure on all instances such as this, the procedure requires that the floor be evacuated.

THE REPORTER: Off the record for a minute.

(Recess taken)

BY MS. BURTON:

Q So, Mr. Jensen, we've gone through a number of events at the Millstone spent fuel pool involving problems with fuel handling. And would you still agree that there may be more that have not been brought to our attention through this discovery process based on all your testimony?

A I think the possibility exists. I don't know of any.

Q If you knew of them, I assume you would have brought them to our attention by now?

A Absolutely.

Q Do you know what the standards are for qualification of fuel handlers?

A Not precisely. There's a training program and there's -- it consists both of classroom training and on-the-job training, and a qualification card is filled out and approved, and the person becomes qualified.

Q The process of fuel handling involves quite a number of personnel, correct?

A Yes.

Q Who is at the top of the hierarchy in terms of directing fuel handling?

A The direction for fuel handling and placement of special nuclear materials all comes from reactor engineering generated forms; either material transfer form or refueling work list.

Q Now, the plant operators who operate the control room, when they are qualified to operate the control room, are they also at the same time qualified to be operators of fuel movement?

A Because a person has an NRC license, RO or SRO and has completed his control room qualifications does not qualify him to operate refueling equipment. That is a separate qualification -- it is -- it may include it, but it's doesn't -- it's not required to be included. It's not part of the NRC's examination process. We hold separate qualifications on that

equipment. Nor do you have to have an NRC license to be qualified as a fuel handler.

Q A fuel handler, would that include somebody who's operating the crane that lowers the fuel?

A It basically is a crane operator qualification, but it's for the fuel handling, correct.

Q Are you familiar with the proceedings that were brought about by the U.S. Department of Justice that led to criminal penalties last September?

A Criminal penalties against Millstone?

Q Against Northeast Nuclear Energy Company.

A You would have to give me more information. I'm not sure what you're talking about.

Q Well, I'm talking about the day when Mr. Michael Morris pleaded guilty to charges under -- felonies under the Atomic Energy Act, and also the Clean Water Act.

A I'm aware that he did plead that, yes.

Q And that the charges included felonies under the Atomic Energy Act involving falsification of training records for operators?

A That was my understanding as to one of the charges, yes.

Q Now, do you know if those charges extended to the qualifications of individuals to work in the spent

fuel pool?

A No, I do not know.

Q Mr. Jensen, I understand that you went along on the site visit to Unit 3 to the spent fuel pool yesterday?

A Yes, I did.

Q And I understand that photographs were taken?

A Yes.

Q Are they available now?

MR. REPKA: They should be available in the next day or so. We just haven't been there today, so I don't know whether they are done.

BY MS. BURTON:

Q Now, I think that it was observed that there are certain pipes overhead of the pool?

A Yes.

Q And, in fact, I think that I understand that there was discussion about a boron dilution analysis that led to certain things to be done to one of the pipes that is overhead of the pool?

A I'm not sure of a boron dilution analysis or anything. We did discuss the pipe above the pool. The pipe is a drain pipe from the roof that was originally designed to carry rain water.

I didn't know its current status, so this

morning I checked, and I was informed that that particular pipe is no longer in service and has been blocked at the roof. In other words, no rain water flows in that pipe currently.

Q When was it blocked?

A I don't have that information, but I can find it.

Q How did you determine that it had been blocked?

A I talked to the spent fuel pool project, in particular, Carl Witiker.

Q Do you have information on how it was blocked?

A No. I was only confirming its current operable status. It is currently not being used, and it's blocked at the roof.

Q Where is the water being diverted now?

A I don't know.

Q Is that an original pipe, drain pipe?

A I don't know. I would assume.

Q And is there an analysis that was done as to the potential for boron dilution attributable to leakage from that pipe?

A I'm not aware. It's possible.

Q Well, if such an analysis were done and you

we were to request it, I assume that you would be able to provide it to us?

A I would have to search for it. It's not an analysis that my group would perform or obtain any copy of. I would have to go to another group.

Q I also understand it was observed in a site visit that there are overhead heating devices?

A Yeah, there's an overhead heating coil and fan.

Q One coil and one fan?

A It's a unit. It's a coil fan unit with supply and return lines.

Q What are the approximate dimensions of it?

A That (indicating).

Q Three feet, four feet?

A Yeah.

Q By?

A Four feet by three feet.

Q By?

A Maybe that thick (indicating) with the fan on.

Q Two feet?

A Eighteen inches.

Q And it's located directly overhead of the pool?

A It's directly over the curb, the eastern-most curb of the pool.

Q And is that in operation?

A I don't know.

Q I don't mean today, but generally?

A I don't even know generally.

Q Are there other pipes that are overhead -- other pipes or devices that could be collectors of water located above the pool?

A There were a couple of lines that ran on the roofing support system, but I don't know what they were. They are --

Q You don't know what they are?

A I don't know what they are. They were silver insulated pipes.

Q Are there pipes along the walls?

A There is -- there are some pipes located on the western-most wall. They also appear to be heating pipes, and there are some closed cooling water pipes on that wall.

Q Are there pipes on the other walls?

A On the northern-most wall, there is a -- there is a hose fire station on the eastern side of the northern wall.

Q There is what?

A A fire -- a hose station. A fire line comes up and there is a coiled hose there.

Q Okay. What about the other walls?

A The western-most wall, the northern end of the western-most wall, a large fire line comes up with an isolation valve and a cap on it. No other pipes on that wall, and there are no pipes on the southern-most wall, to my recollection.

Q Are you familiar with any events at Units 2 or 3 where there has been inadvertent leakage through a valve that was mispositioned leading to a drop in the level of water in the pool that went undetected for a significant period of time?

A None that went undetected for a significant period of time.

Q Any that went undetected at all?

A None that went undetected at all.

Q Have there been any leakages from either the Unit 2 or 3 pools through the fact of malpositioning of valves?

A I'm unaware of any.

Q Do you have any familiarity with the Institute for Nuclear Power Operations?

A I have some familiarity in areas.

Q Do you know if Millstone or its operators is

a member of INPO?

A Northeast Utilities is a member of INPO.

Q Do you know if Northeast Utilities has data concerning industry-wide experience in boron dilution fuel mishandling in spent fuel pools?

A Northeast Utilities has access electronically to a couple of the different databases that INPO supplies; one of them being Operating Experience Reports, and we can do searches on that database, yes.

Q Is there information on the database pertinent to industry-wide boron dilutions or actual mishandling in spent fuel pool?

A I don't know. I personally have not searched under that query.

Q Are you familiar with the process of fuel handling, the movement of fuel at the spent fuel pools?

A Yes.

Q Is there a computerized component to the process?

A I guess it would depend on what you define as "the process." We have a procedure that develops and implements fuel movements. That process is all hand calculated, handwritten. And we do use a program that we purchased from Combustion Engineering, now it's ABB,

called Shuffle Works. We use that as a tool to aid us in fuel movements.

However, it's not procedurally required. It's not something that we're required to use. We use it because of its ease of tracking fuel moves. It also has routines in it that can check errors and things like that, so it's only used as a check tool, it's not used formally as part of the process.

Q Do you know if it is possible to know in realtime where each fuel assembly is at all times?

A In -- yes. We have material transfer forms, and those material transfer forms dictate what fuel is to be moved where. That, in conjunction with SNM card file. The difference being the SNM card file is organized by component by each piece of special nuclear material. And a material transfer list is organized by the sequence of the different moves.

If you have completed a sequence of moves of special nuclear material, the next step in the process is to update the cards, the SNM cards.

Q What is the lag time?

A The lag time is typically two to three weeks.

Q And that would be between the time that the actual movement is made and the information --

A Index cards are updated, yes, ma'am.

Q So there could be a period of two to three weeks when, typically, the information is not current as to where the fuel bundles are located, fuel assemblies?

A The information on those cards may not be current, but my group has the current information. As I said, all special nuclear material movements are controlled by my group, and only my group. The material transfer forms and the refuel work lists are generated and controlled by my group, and we're the group that updates the cards.

Q Do you know if there have been any License Event Reports filed concerning the Millstone operations at Units 2 and 3 since they were restarted in 1988 and 1999?

A I'm aware that there have been some, yes.

Q Can you identify them?

A Not off the top of my head, no.

Q Do any concern the spent fuel pools?

A I can't remember.

Q Do any of them concern administrative controls?

A That I don't know.

Q If we were to ask you to look up that information, you would probably be able to provide it

to us?

A For LER's, absolutely.

MR. REPKA: That's something you could do as well off the NRC's database.

THE WITNESS: Or in a public document room.

BY MS. BURTON:

Q Now, I understand that you assumed a role during the site visit yesterday to the spent fuel pool of providing information. Was that formal or informal?

A It was informal. I would probably categorize it as a tour guide.

Q Could you tell me if anything -- any special maintenance was done to the pool, or if any changes were made that were not scheduled prior to the visit?

A You mean did we do anything special for the visit?

Q Yes.

A No.

Q Was there any chemical change that was -- no special chemistry was applied?

A No.

Q Has the lighting at Millstone 3 been changed at all since the plant went on line in 1986?

A Yes.

Q How so?

A We've had lights go out, and we've had to replace them. We move lights around, and we added a couple of lights in the spent fuel pool.

Q Where?

A They are movable, so they can be at any point. Again, they hang from the curb, and I can move them wherever I like them to support the work activity.

Q So additional lighting has been installed at the Unit 3 spent fuel pool?

A Since startup, yes.

Q When?

A I would have to look up the dates.

Q Recently, during your personal experience there?

A The only thing we've done in the last two years is relamp the existing lighting.

Q By "relamp," you mean --

A Replace burned out light bulbs.

Q Uh-huh. Within the past two years?

A Yes.

Q And that's Unit 3?

A Both Units 2 and 3 we've done.

Q Just replacing?

A Just replacing burned out light bulbs. It's

kind of a big deal. One, the bulbs are very expensive, and they have to be sealed up because they are under water.

Q How expensive is that?

A I think they run in the neighborhood of about -- just the lamp itself is just under $2,000.

Q And how many lamps -- are we talking Unit 2 or Unit 3?

A They are roughly equivalent in price.

Q And how many lamps of that description are there in each of those pools?

A I believe currently I have six lamps in operation in the Unit 2 spent fuel pool, and I can't remember Unit 3. The -- we're in a refueling outage for Unit 2, so I have the pool completely lit up with all the lamps.

In Unit 3, we're not in a refueling outage, so the ones in the transfer canal I have turned off, so I can't remember exactly how many I have. I only have the ones in the pool itself illuminated, and I think there's four or five.

Q Now, when these bulbs go out, they are not automatically replaced?

A Because it's -- it's a fairly long process, it involves the removing of a potentially radioactive

component out of the spent fuel pool, the lights themselves are fairly expensive, the replacement lights, if we haven't had a need for having that many lights there, then no, we don't replace them right away, we replace two or three at one time.

Q I'm just trying to understand the sequence here. You said that in the past two years, lights have been replaced?

A Yes.

Q What is the longest period of time between replacements of bulbs that have burned out?

A I don't know.

Q Not two years?

A Again, that predates me. Well, it could be. The reason the lamps are so expensive is because they are high lumen long-life lamps. They typically can be illuminated for five to ten years without burning out. So we can have one or two go out in a four or five-year period and not do anything about it, and then just before we refuel when we have activities in the fuel pool, we will, in fact, relamp them all, all the ones that are burned out.

Q But you say there have been occasions when lights have been out for as long as four or five years?

A I'm saying that's possible. I don't have an

exact number for a duration of a particular lamp being out.

Q So in terms of lightage, you have six of these big lamps at Unit 2. What other lights in addition to these $2,000 units?

A Well, there's the overhead building lamps. Again, these are ones -- these particular lights we're talking about are on long, high polished poles. And they are high polished so they don't -- things don't adhere to them, and it's easier to decontaminate should it be needed.

They come down, there's a ballast that sits on them, and then a lower pole, there's a reflector unit that sits on them, and they sit inside that, and they hang off the curb. Those are the lamps we're talking about. There are six of them in the Unit 2 spent fuel pool right now.

Now, the pool exists within the building, and the building has lights within the building, and I believe they are high efficiency sodium lamps. And they do provide some lighting, but not direct lighting. And we do have the capability to put drop lights if we have a particular area we want to illuminate.

Q Are you familiar with the violation recently issued by the Nuclear Regulatory Commission against

Northeast Utilities concerning alteration of a safety document characterized by the New London Day as in an attempt to cover up mistakes?

A No, I'm not familiar with it.

Q I'd like to show you a newspaper article and see if that will refresh your recollection. Does that refresh your recollection?

A Well, I have no personal knowledge of it, other than the newspaper article.

Q Had you seen it before? Were you aware of it before?

A Only by title, that, you know, office conversation, hey, there was this issue. Okay.

Q Going back to what we were mentioning earlier about the criminal sanctions for violations under the Atomic Energy Act for falsifying training records --

A Yeah.

Q -- are you familiar with the particular individuals involved, who it was alleged had not completed proper training before they were certified to operate the plants?

A I'm familiar with the Unit 1 operational staff, and as such, I'm probably familiar with those people, yes.

Q It was all Unit 1?

A I believe that -- well, I'm not sure, but I do know that some of the contentions involved Unit 1.

Q Now -- and the individuals involved you're associating with Unit 1?

A It was my understanding that the problems with records occurred in the operator licensing branch, and I'm familiar with all of the personnel in the operations department. So by virtue of that, am I familiar with the persons involved, I would have to say yes. But I don't know who or what constituted the violation.

Q Well, do you know the individuals involved whose training problems gave rise to these precedent setting, I understand, penalties under the Atomic Energy Act, and are they still working at Millstone?

A I -- by virtue of the fact I know everybody in the operations department, I have to say I know the individuals, who those individuals are. I don't know, so I can't say that they still work there or not.

Q So do you have any information as far as who the individuals were who were the subject of the criminal felonies?

A Not specifically, no.

Q You mentioned something --

MR. REPKA: I think you're assuming something here. You're assuming criminal penalties went to the operators as opposed to the trainers.

MS. BURTON: No, I'm not assuming that.

MR. REPKA: I think you're creating that impression, and I think it's inaccurate.

MS. BURTON: The penalties were paid by the company.

MR. REPKA: I understand that.

MS. BURTON: Right.

MR. REPKA: But the misconduct, you're focusing on operators, but I wouldn't assume that the misconduct was on the part of the operators.

MS. BURTON: I wasn't assuming that at all.

THE WITNESS: Okay.

BY MS. BURTON:

Q I'm just asking, Mr. Jensen, if you happen to be familiar with any of the individuals whose training records were the subject of the federal action?

A Here's what I know: I know that there is an allegation of training record falsification that occurred within the company and apparently was substantiated. It involved operators, and I know all the operators, but I do not know the links between the

two. So I don't know who in the operations department it involved or what actually occurred as far as what constituted the falsification, so --

Q Do you know if there are any fewer operators today, or if any of the operators that you were aware of at Millstone at the time of the criminal penalties being imposed, if any of them have left, or if they are all still there?

A They are not all still there. Millstone Unit 1 has entered a decommissioning stage, and as such, they no longer have licensed operators. They have what they call certified fuel operators. And as such, the operations staff has significantly shrunk. They were down to 30, 40 percent if the plant were operating, staff size.

Q Did some of the people who were at Unit 1 transfer over to Units 2 and 3?

A Yes, they did.

Q Including some operators?

A Yes.

Q And with regard to the penalties under the Clean Water Act, are you familiar at all with the allegations concerning willful, false sampling of environmental discharges?

A I understand that is an allegation. I have

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	:	Docket No. 50-423-LA-3
Northeast Nuclear Energy Company	:	
Millstone Nuclear Power Station, Unit No. 3	:	MAY 11, 2000

DEPOSITION OF MICHAEL C. JENSEN

_____

MICHAEL C. JENSEN

Subscribed and sworn to before me this _____ day of _____, 2000.

_____

Notary Public

My Commission Expires:

STATE OF CONNECTICUT)
)
COUNTY OF NEW LONDON)

I, Kathryn Orofino, a Notary Public within and for the State of Connecticut, do hereby certify that I took the deposition of MICHAEL C. JENSEN, a witness above-entitled action pursuant to 10 CFR Section 2.740a on the 11th day of May, 2000, at the Mystic-Noank Library, 40 Library Street, Mystic, Connecticut, at 1:40 p.m.

I further certify that said witness was by me duly sworn to testify to the truth, the whole truth and nothing but the truth, and that the testimony was taken by me stenographically and thereafter reduced to writing under my supervision; and that I am not an attorney, relative or employee of any party hereto nor otherwise interested in the event of this cause.

In witness whereof, I have hereunto set my hand and affixed my seal this 30th day of May, 2000.

Kathryn Orofino
Shorthand Reporter #342
Notary Public

My Notary Public Commission Expires March 31st, 2001

EXHIBIT 11

Matthew L. Wald, The New York Times, June 30, 2000, Page B1 ("Con Ed Put Off Plant Upgrade Over Rate Fear")

The New York Times
June 30, 2000
Page B1

Con Ed Put Off Plant Upgrade Over Rate Fear

Relied on Faulty Report Of Safety at Indian Pt.

By MATTHEW L. WALD

Consolidated Edison decided in 1997 not to replace the steam generator that would cause an accident at a Westchester County nuclear reactor two and a half years later because the company was uncertain whether the move was a good financial bet in the deregulated market that was developing, according to an internal planning document.

Some utility industry experts say the document may be the first evidence that electricity deregulation can compromise nuclear safety, a concern that critics have voiced for years.

The accident, on Feb. 15 at Con Ed's Indian Point 2 nuclear reactor in Buchanan, N.Y., was the most serious in the reactor's 27-year history. A small amount of radioactive steam escaped after corrosion cracked a tube in one of the reactor's four steam generators, which carry superheated radioactive water.

While no one was hurt and Con Edison says the amount of radiation released was tiny, the accident has had serious consequences, including the shutting of the plant for at least five months, and possibly longer, at a time of tight electricity supplies. It has also complicated the company's efforts to sell the reactor.

In October 1997, Con Ed financial planners concluded that replacing the reactor's steam generators soon was the cheapest option for customers and shareholders. Their analysis noted that the generators were deteriorating — a common occurrence in reactors — limiting how much electricity they could produce. And if the generators were not replaced, they would have to be inspected more often, cutting the number of days the plant could run, according to the planners' document, which was provided to The New York Times by Edward A. Smeloff, a utility expert at Pace University Law School who has been critical of Con Ed's performance in running the reactor.

But Con Ed's analysis also pointed out that its financial projections were highly sensitive to the price of electricity and that postponing a decision would give the company an opportunity to refine its estimates as

Continued on Page B5

Con Ed Put Off Upgrading Indian Pt. Over Rate Fears

Continued From Page B1

the state made its transition to a deregulated electricity market. That transformation happened last November.

In their analysis, the financial planners accepted a judgment — which turned out to be wrong — by Con Ed engineers that the existing steam generators were safe for continued use, although if kept in place they would need an extra inspection each year. As it turned out, Con Ed got permission to skip the extra inspection in 1999; it would have been the last one before the accident.

Asked about the analysis, a vice president of Con Edison, Steven E. Quinn, said yesterday that the benefit projected for replacing the steam generators — $85 million over 14 years — was too small to justify the financial risk, because the uncertainties were so large. He said, though, that those uncertainties were not just the future cost of power but also how well the plant would run after the replacement.

"The uncertainty on the assumptions was large," he said.

The Con Ed analysis compared three options for the reactor: replacing the steam generators and running the plant until its license expired in 2013; not replacing the generators and running the plant until 2013, but at a lower power level and with an extra shutdown every year for inspections, averaging 30 to 36 days; or simply retiring the plant in 1999 or 2001. The first option was judged the least expensive.

Mr. Smeloff, the director of the Pace Law School Energy Project and a former utility manager, said in a telephone interview: "Even from a shareholder perspective, replacing steam generators in '99 made economic sense. If you assume management was acting in the best interest of shareholders, this is the choice they would have made."

But King Look, a section manager in Con Edison's generation planning department and one of the authors of the document, said the problem was that the financial projections were highly sensitive to electricity prices, and that no one knew how those prices would run in a deregulated market.

Con Ed projected that replacing the steam generators would cost $121 million, not including the cost of the equipment itself. Con Ed has replacement generators on site, which it obtained from Westinghouse, the original manufacturer, as part of a legal settlement in the 1990's.

The company figured that the cost of running the plant until license expiration in 2013 was $1.52 billion; shutting it down in 1999 would cost $59 million more, including replacement power costs, but replacing the steam generators would save $85 million.

The projections were of "net present value," a common technique in business analysis that means taking interest rates into account and valuing a dollar today more than a dollar a year from now. They assumed an extra annual shutdown for steam generator inspection, and assumed that with new steam generators, the plant's maximum power level could rise 30 megawatts, about 3.5 percent.

The fear that deregulation may compromise reactor safety has often been voiced but, experts say, seldom if ever borne out. In 1994, Ivan Selin, then chairman of the Nuclear Regulatory Commission, reacting to nascent signs of deregulation in California, told reporters that "even financially sound utilities are under great pressure to reduce their rates, to be competitive; they may be tempted to put off capital investment that we consider necessary to maintain equipment in top shape."

Con Edison asked the Nuclear Regulatory Commission in June for permission to restart the plant with the existing steam generators and run it for up to 10 months without reinspection, although the company now says it will replace the steam generators later this year. The commission is expected to rule next month.

EXHIBIT 12

Memorandum of J.F. Beaupre (NNECO) to D.E. Anderson (NNECO)(June 24, 1999)

Northeast
Utilities System

CR M3-99-2236 3 Pages

Memo

To: D. E. Andersen
N. G. Bergh
D. C. Gorence
Nuclear Oversight

June 24, 1999
MP3-TS-99-185

From: J. F. Beaupre
Unit 3 Technical Support Engineering

Title: Response to Audit Finding, CR-M3-2236, "Adverse Trend in Performance of the Refueling Equipment"

SUMMARY

During RFO6 core offload and onload, the fuel handling system experienced numerous and varied equipment failures which resulted in delays to the refueling schedule. Although these equipment failures did not result in actual fuel damage, the number and variety of failures demonstrated that the fuel handling system was not adequately prepared to support refueling operations. This memorandum summarizes the fuel handling system equipment failures that occurred during RFO6 and corrective actions that have been completed, lists the apparent causes for the failures and provides corrective actions to assure the equipment will be ready to operate reliably in future refueling outages.

EQUIPMENT FAILURES AND REPAIRS

The significant equipment failures that occurred during fuel movement are:

1. The fuel transfer cart had difficulty traversing the final few inches to the fuel pool upender. The cart would frequently stop approximately ¼ inch from the end stop and this prevented one or both of the cart locking blocks from engaging when the fuel basket was raised. Whenever both blocks failed to engage, the traverse drive motor torque switch would reset and an interlock in the upender control circuit would then prevent the basket from lowering back to a horizontal position. After core offload, personnel identified that the cart holddown latch springs were binding and stopping the cart from travelling to the end stop. These springs were replaced with an improved design, however, mechanics also discovered that the cart is rubbing on the tracks during the last few inches of travel into the fuel building. During core onload, this condition improved considerably but further work is required to eliminate the rubbing.
2. The SIGMA refueling machine gripper and stop plate limit switch cable failed, resulting in intermittent problems while latching and unlatching fuel assemblies in the core and at the upender. Technicians suspected that a connector on the cable had failed. This connector had been installed during RFO5 because the cable supplied by Westinghouse for a mast modification was too short and an additional length of cable was needed. After a few time-consuming and unsuccessful attempts to repair the connector, the entire cable was replaced. The cable replacement eliminated the problem.

3. The fuel transfer cart holddown latch failed to return to center when the cart left the fuel building end stop. This failure was initially attributed to the jammed springs that were replaced, however, the problem still existed during the onload, and further investigation is required.
4. The spent fuel bridge hoist manual drive chain became misaligned with the tensioner sprocket while raising a fuel assembly from the upender. This caused the hoist to stop and required the crane operator to lower the fuel assembly back into the upender. After unlatching the tool, the hoist again stopped before the tool was above the top of the basket. The tool was lowered at the minimum hoist speed and subsequently raised sufficiently to clear the basket. After placing the tool in its storage bracket, the manual drive chain and sprockets were removed under a temporary modification. The hoist operated reliably for the remainder of the refueling.
5. While closing the fuel transfer tube gate valve, the reach rod slipped down in its support and prevented the PEO from fully closing the valve. The reach rod was repositioned and subsequently cycled in both directions with no problems.
6. The communications system for the refueling stations (i.e Control Room, SIGMA and Spent Fuel Building) was unreliable.
7. The SIGMA refueling machine frequently needed to be reinitialized after jogging small distances because the control system does not register these movements correctly. An upgrade to the positioning system is needed to solve this problem.

APPARENT CAUSES

1. Corrective actions to resolve previously-identified fuel handling system equipment problems are frequently ineffective. The SIGMA control problems were identified in RFO4, yet an EWR to upgrade the control system was not scheduled for implementation until Cycle 7. When the SIGMA cable supplied with a mast modification was identified as being too short, an effort to replace the cable with the proper length should have been initiated. An EWR to replace the spent fuel bridge hoist manual chain drive with a simpler design was approved, but the design change was given low priority and not completed prior to RFO6. The transfer cart holddown latch was modified after RFO1, yet failed to operate properly during RFO5 and RFO6. Efforts to repair the latch during RFO5 were unsuccessful. The new transfer cart holddown latch springs appear to be too weak to overcome friction in the latch bushing and return the latch to center. The transfer tube gate valve reach rod had slipped down during RFO5 and a modification to the support was not fully effective. Problems with the communications system were identified in RFO5 and were not effectively resolved prior to RFO6.
2. Operating experience at other plants is not effectively evaluated for applicability at Unit 3 and incorporated into the preventative maintenance program. Fuel handling system vendor manuals state that the equipment was designed to be reliable and the manuals specify the maintenance that needs to be performed prior to refueling outages. However, experience has shown that performing the minimum recommended maintenance does not assure good performance. As the equipment ages, unanticipated failures have occurred. Thoroughly reviewing fuel handling system problems that have occurred at other plants provides a foundation for evaluating the adequacies of Unit 3's PM program.
3. Preparing the fuel handling system for refueling is given low priority while the plant is online. Preventative maintenance which is scheduled months before the outage is frequently deferred to a later start date because of other priorities. This results in significant pressure to complete the fuel handling system PMs in a short time, immediately prior to the outage. The consequences of delaying the PMs is that problems identified must be corrected quickly and this sometimes results in the ineffective corrective actions previously identified.

4. Failures of fuel handling system equipment that delay refueling are not perceived to be safety-significant. This is demonstrated by the EWR prioritization process that assigns point values to EWRs based on significance (i.e. safety, cost-savings, ALARA, etc.). A review of EWRs related to the reliability of the fuel handling equipment shows that the safety significance of equipment upgrades is not fully understood and communicated to management.

CORRECTIVE ACTIONS

To provide assurance that the fuel handling system performs reliably in future refueling outages, the following corrective actions will be performed:

1. Evaluate potential PM program enhancements based on reviews of the following:
   a. ANSI requirements for crane inspections.
   b. PMs recommended by OEMs.
   c. Open AWOs on fuel handling system components.
   d. CRs previously written against fuel handling system.
   e. Refuel team and Reactor Engineering logs.
   f. Historical fuel handling system corrective maintenance AWOs.
   g. New and previously-evaluated refueling equipment lessons learned.
   h. Industry OE for fuel handling equipment.
2. Visit fuel handling equipment vendors and selected plants to evaluate the design and performance capabilities of potential upgrades to the fuel handling system.
3. At least 15 months prior to RFO7, recommend upgrades for fuel handling system to management via EWR process.
4. At least 12 months prior to RFO7, establish a schedule to complete all fuel handling system DC, PM and CM AWOs prior to core offload.
5. At least 6 months prior to RFO7, review all procedures containing preoperational testing requirements and recommend enhancements where desired.
6. At least 3 months prior to RFO7, complete a Technical Evaluation of refueling equipment readiness.
7. Perform an effectiveness review of these corrective actions following RFO7.

c: P. B. Dillon
V. P. Spunar
G. L. Swider

EXHIBIT 13

Letter of James C. Linville (NRC) to R.P. Necci (NNECO) (July 9, 1999)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

July 9, 1999

#8
#45

Mr. R. P. Necci, Vice President
Nuclear Oversight and Regulatory Affairs
c/o Mr. D. A. Smith, Manager - Regulatory Affairs
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, Connecticut 06385

SUBJECT: NRC COMBINED INSPECTION 50-336/99-06 and 50-423/99-06

Dear Mr. Necci:

On June 14, 1999, the NRC completed an inspection at Millstone Units 2 & 3 reactor facilities. The enclosed report presents the results of that inspection.

During the eight-week period covered by this inspection period, your conduct of activities at the Millstone facilities was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls.

As documented in the enclosed report, we focused our attention to Unit 2 operations throughout the inspection period. Specifically, we conducted sustained inspections of control room activities from reactor criticality through the power ascension to stable operation at full power. You performed the Unit 2 startup and power ascension in a controlled and conservative manner following a shutdown which lasted in excess of three years. Operators performed evolutions slowly and deliberately and executed the power ascension without any significant events. Although communication between operators was a strength, one area that warrants further attention involves examples of poor communication between operators and other work groups that led to plant configuration changes without operator knowledge. In addition, during a pre-job brief an operator identified an inadequate surveillance for the atmospheric dump valves which if performed as written could have resulted in a reactor trip. Although it is good that operators are properly addressing these procedural issues as they arise, reliance on individuals performing the procedures to identify procedural deficiencies presents an unnecessary challenge to plant personnel. Line management and nuclear oversight maintained a strong presence in the control room and provided a positive influence on the conduct of operations. In addition to the initial startup, we also observed good operator performance following the May 25, 1999, manual reactor trip and subsequent restart. We will continue to assess your at-power performance with a focus on safety and conservative decision making.

Refueling outage activities were in progress at Unit 3 during most of this inspection period. We observed that the challenges that were encountered during RFO6 were methodically evaluated and appropriately dispositioned by your staff using a team approach. This is generally reflected in the conclusions documented in the enclosed inspection report and in the fact that no new inspection items have been opened. However, we also noted that a number of problems in configuration and work control were either self-identified or self-revealed during this period. Your increased management focus on such concerns addressed the need for more rigorous

process controls on certain tagging and system restoration activities. We understand that your staff is developing longer-term corrective actions to reinforce station management's configuration control expectations and ensure that such events are not repetitive and do not result in more severe consequences.

Based on the results of this inspection, the NRC has determined that 10 Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Appendix C of the Enforcement Policy. These NCVs are described in the subject inspection report. While most of the NCVs involve historical issues, two items are more recent and thus represent more current performance issues. A Unit 2, NRC-identified violation involved the failure to perform design reviews of temporary modifications that were installed through plant procedures. The Unit 3 item, while identified by licensee staff with evidence of effective short term corrective action, involved two separate incidents of a violation of high radiation area requirements. If you contest the violation or severity level of these NCVs , you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with a copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Millstone facility..

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

James C. Linville, Acting Director
Millstone Inspection Staff
Office of the Regional Administrator

Docket Nos. 50-336 and 50-423

Enclosure: NRC Combined Inspection Report 50-336/99-06 and 50-423/99-06

EXHIBIT 14

Intervenors' Interrogatory A2 of Third Set of Interrogatories Directed to NNECO (May 18, 2000)

A2 Boron Dilution

Explanatory Note: The Intervenors seek to identify and characterize scenarios in which the concentration of soluble boron in the Millstone 3 spent fuel pool is reduced through dilution. To that end, the Intervenors seek information about all systems and mechanisms that could add water to the pool or remove water from the pool. Specific questions follow.

(1) Please identify all boron dilution analyses performed for this pool, and provide copies of relevant documents.

(2) Please identify and describe in detail all actions (including backfits and procedural changes) that have been taken to reduce the potential for boron dilution at this pool. Please provide copies of relevant documents.

(3) Please identify and describe in detail all piping and systems that could remove water from this pool and from the pool cooling and purification systems. For the purposes of this question, include all water removal pathways, not only those pathways allowed by present procedures. Please provide diagrams, drawings and specifications of relevant piping and systems.

(4) Please identify and describe the potential effect on the pool water inventory of ruptured or broken tubes in a pool cooling heat exchanger. Please provide relevant documents.

(5) Please identify and describe the potential effect on the pool water inventory of pipe leaks, pump seal leaks, inadvertent opening of drain valves, or other water loss pathways from the pool cooling and purification systems. Please provide relevant documents.

(6) Please identify and describe in detail all piping and systems that could add water to this pool and to the pool cooling and purification systems. For the purposes of this section, include all water addition pathways, not only those pathways allowed by present procedures. Please provide diagrams, drawings and specifications of relevant piping and systems.

(7) Please identify and describe in detail all piping that passes through the pool building that could, through leakage, opening of a valve or flange, or addition of couplings, hoses or spool pieces, cause a flow of water into the pool. Please provide diagrams, drawings and specifications of relevant piping and systems.

(8) Please provide the volumes of the fuel pool, the cask pit, the transfer canal and the reactor refueling cavity.

(9) Please describe the rainwater flow paths on and in the vicinity of the roof of the fuel pool building and provide estimates of rainwater flow volumes.

A3 Design Codes

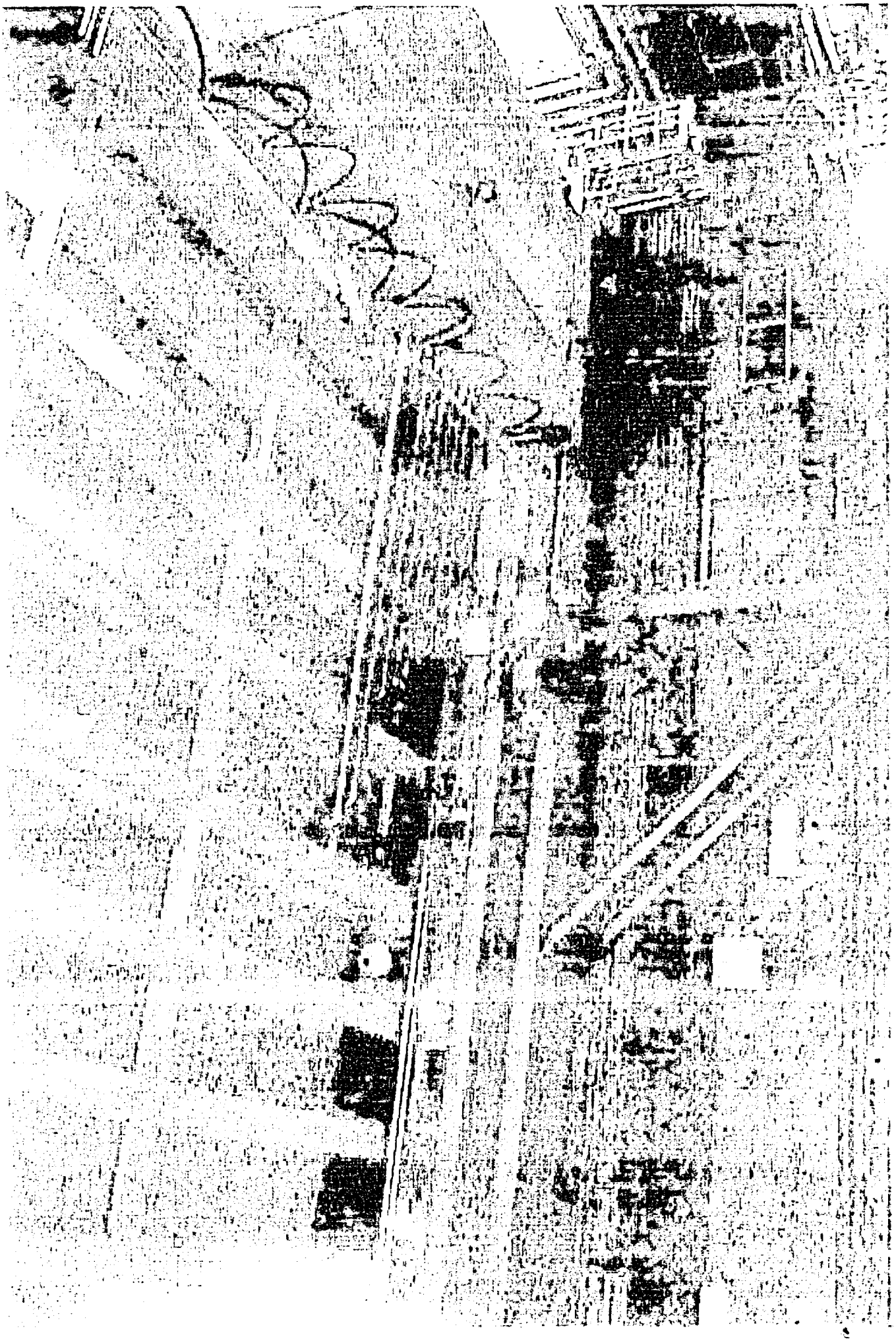
(1) Attachment 5 to the NNECO license amendment application contains Section 2.3 on Codes, Standards and Practices. At page 2-3, this Section lists the design code ANSI N210-1976. The American Nuclear Society has revised this code and has incorporated the revision in the code ANSI/ANS-57.2-1983. Is NNECO bound by ANSI/ANS-57.2-1983 for the purposes of the requested license amendment?

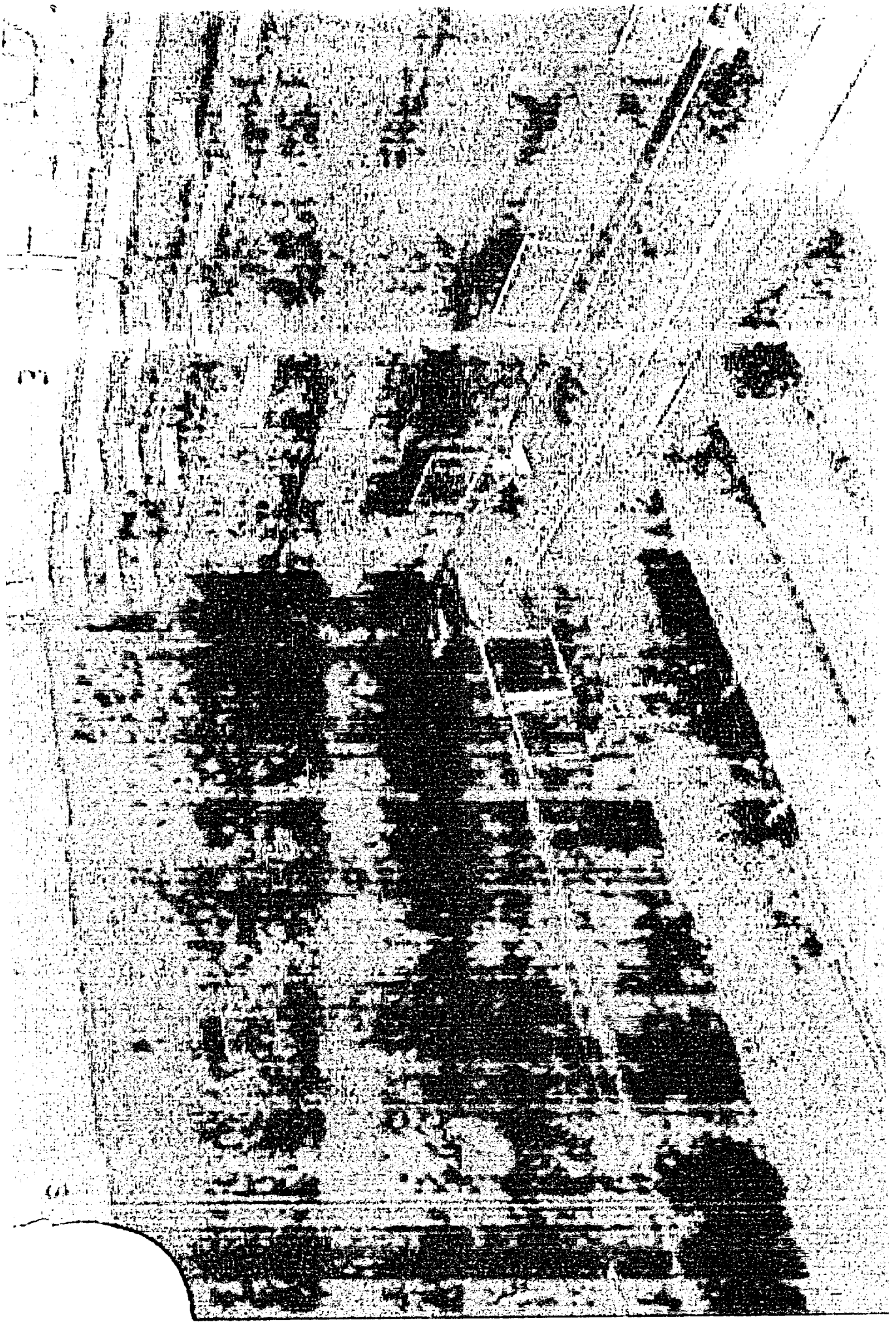
A4 Calculations of K-EFF

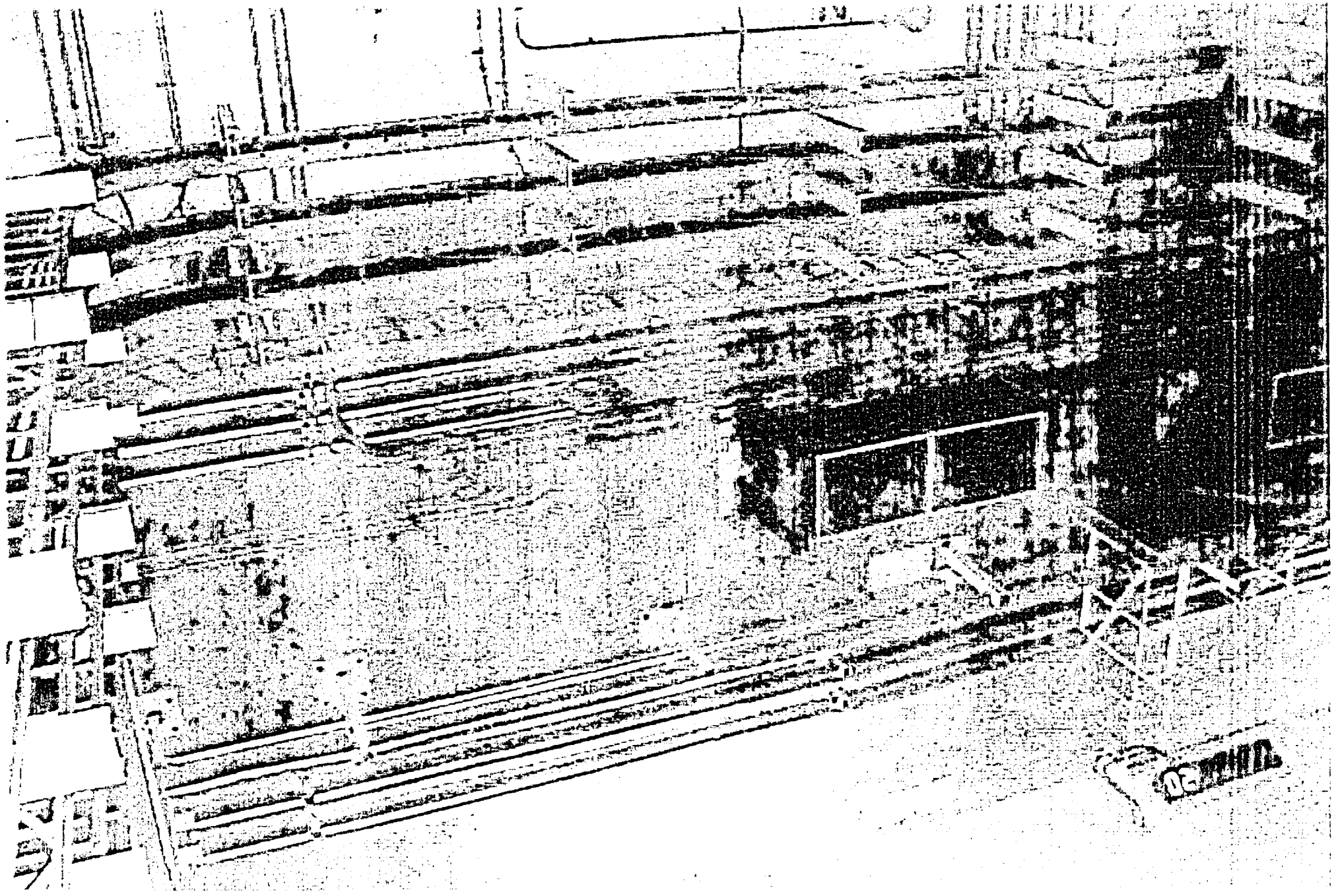
(1) Given the implementation of the proposed re-racking of the Millstone 3 pool, and assuming an absence of soluble boron, what would be the calculated K-effective in each of the regions of the pool if various combinations of fresh fuel assemblies were placed in

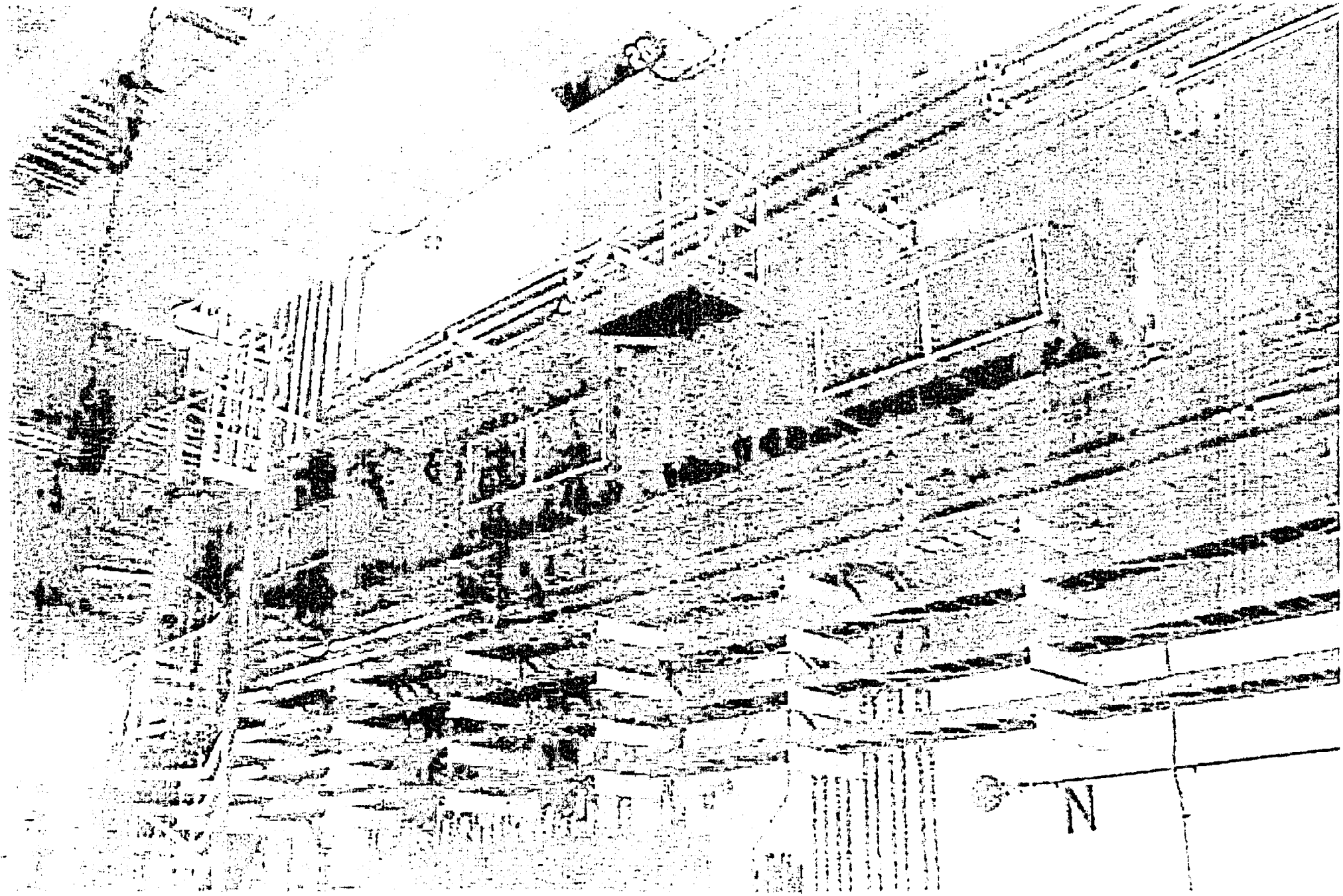
EXHIBIT 15

Set of Photographs of Millstone Unit 3 Spent Fuel Pool Provided By NNECO





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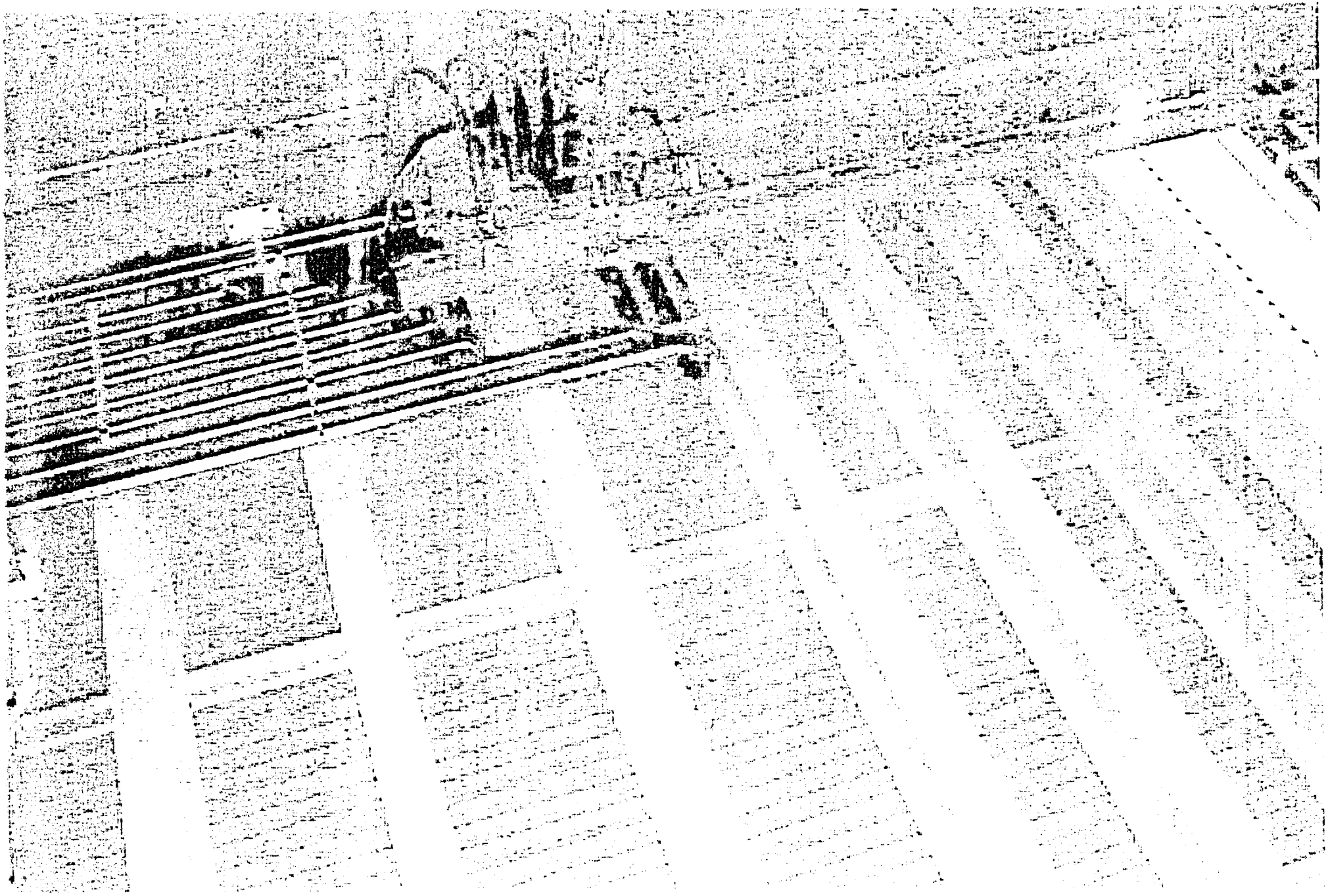

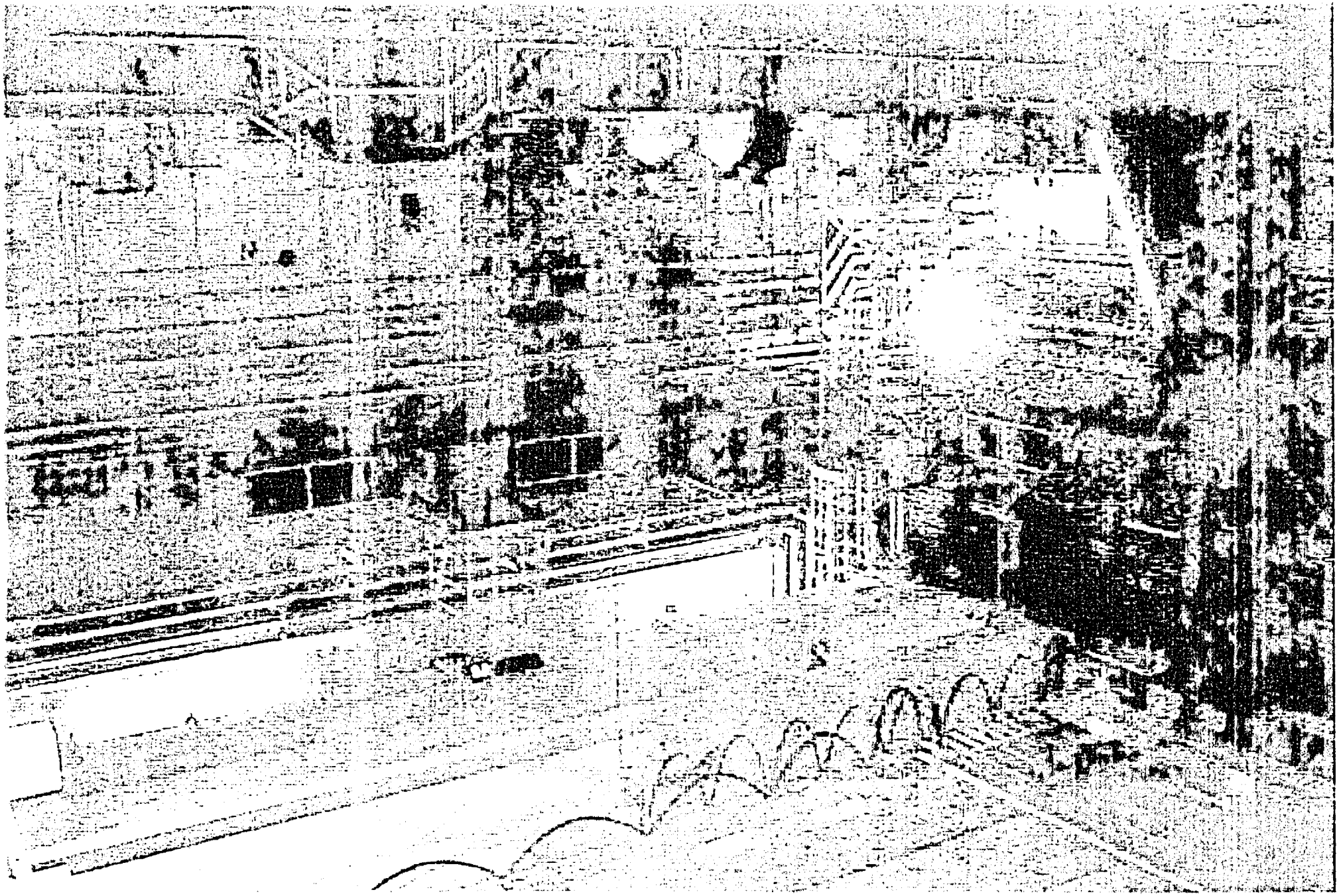

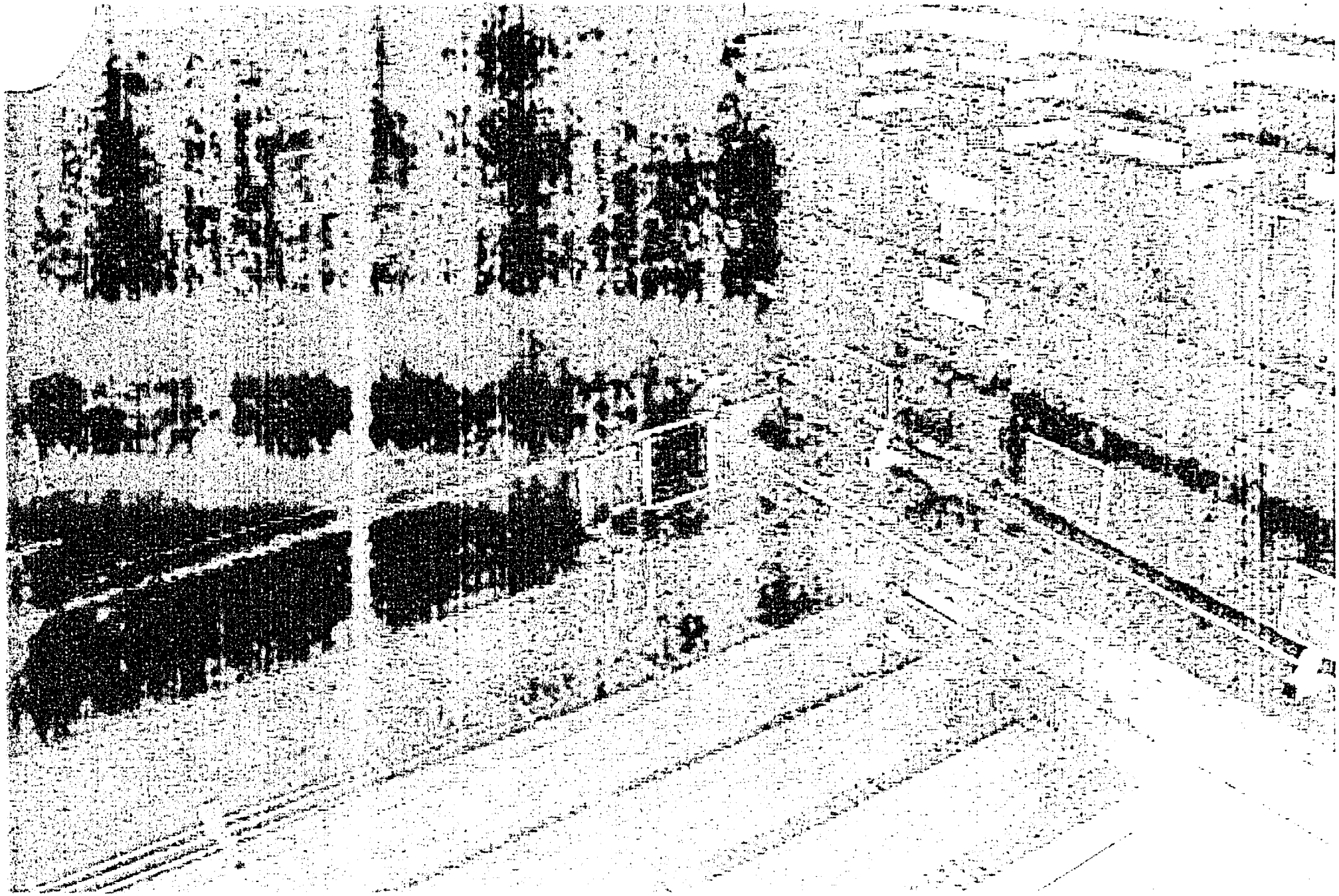

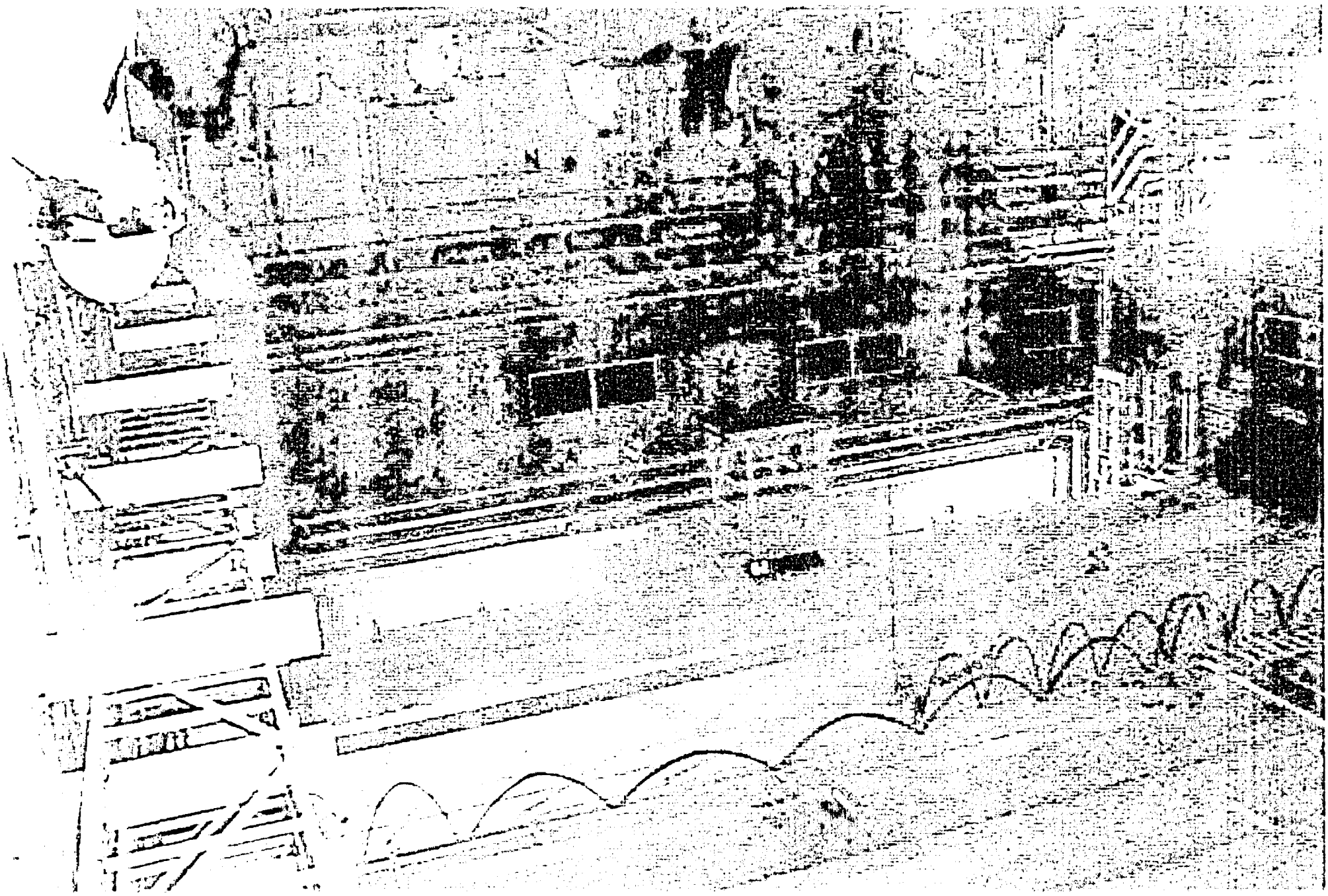

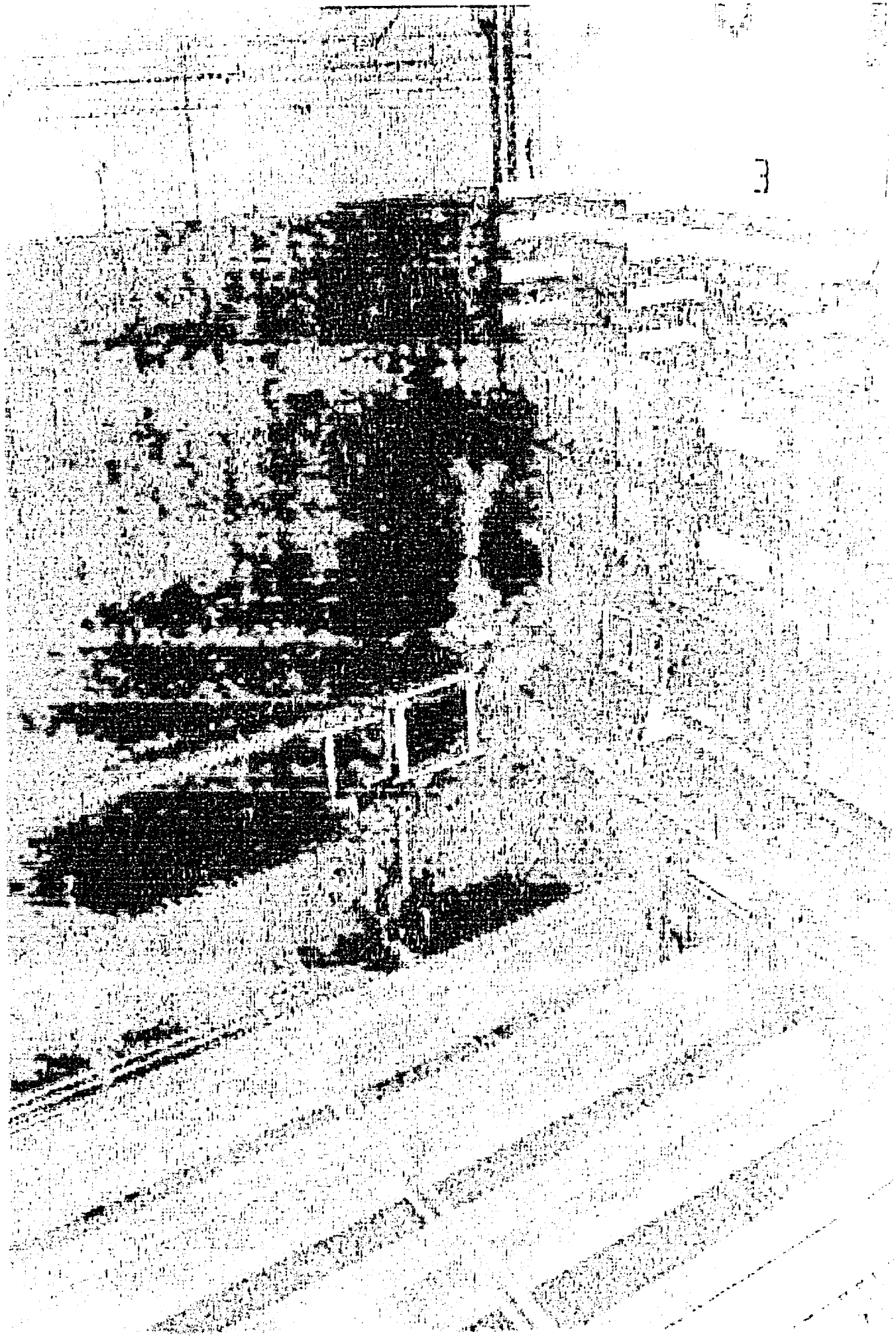

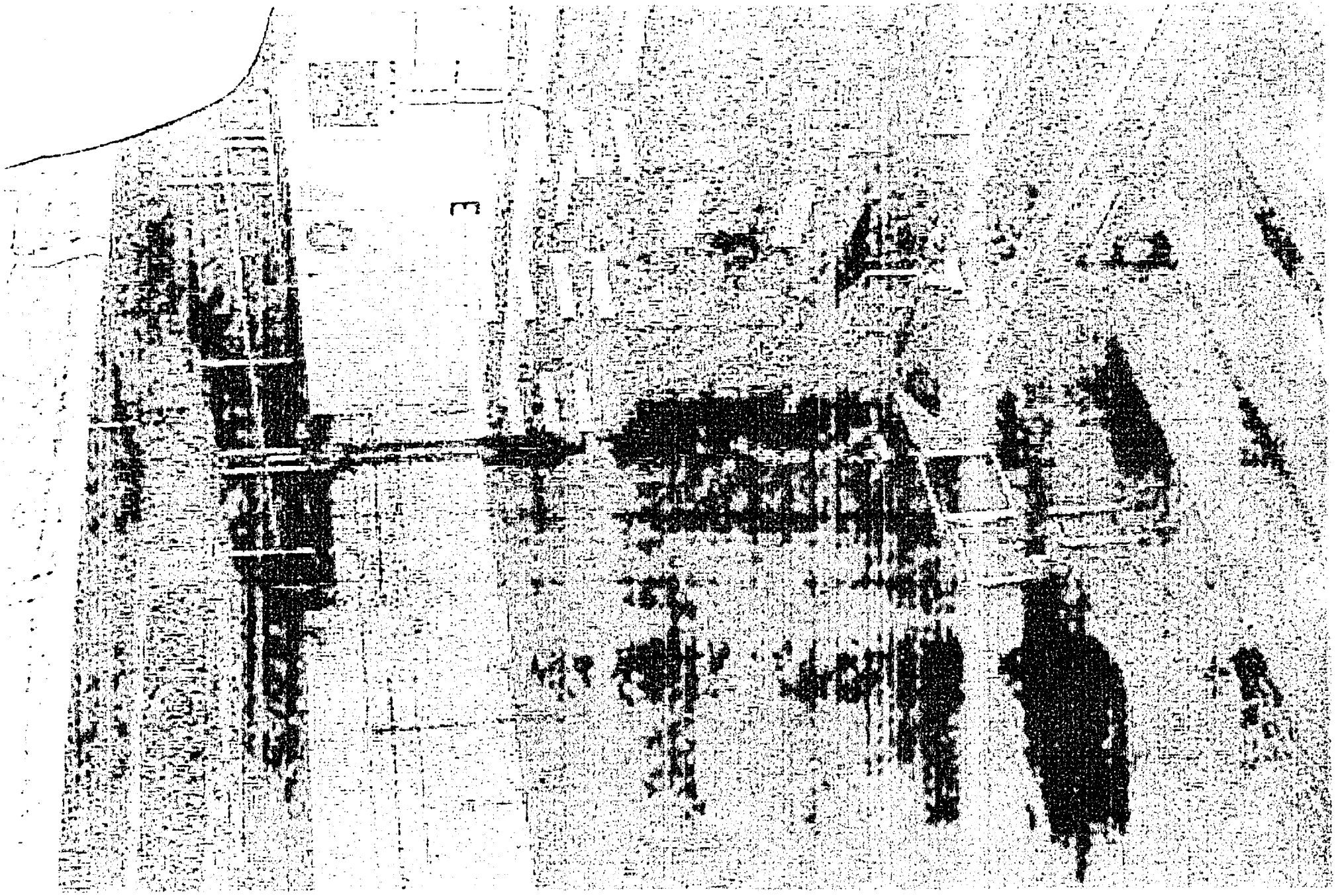

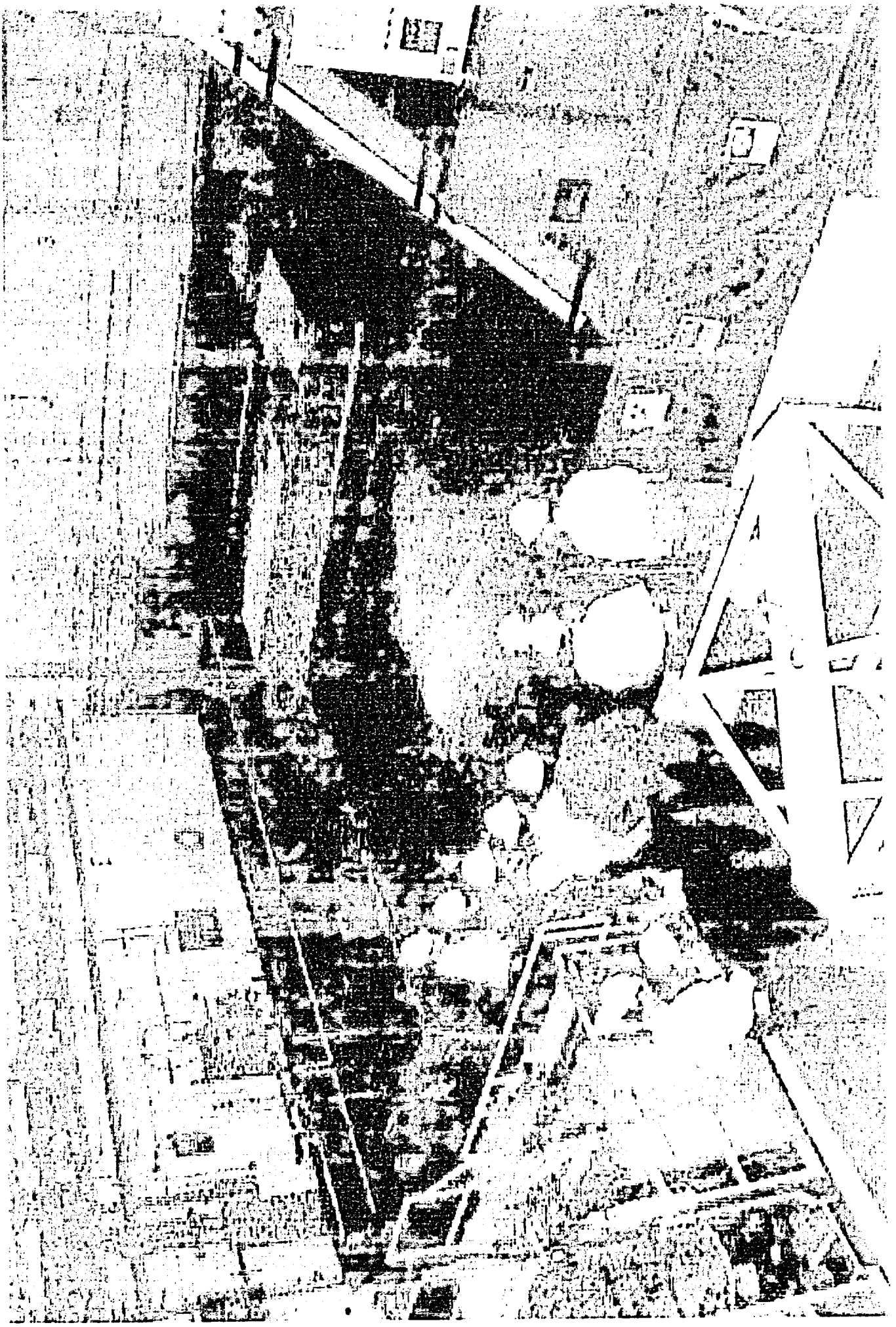

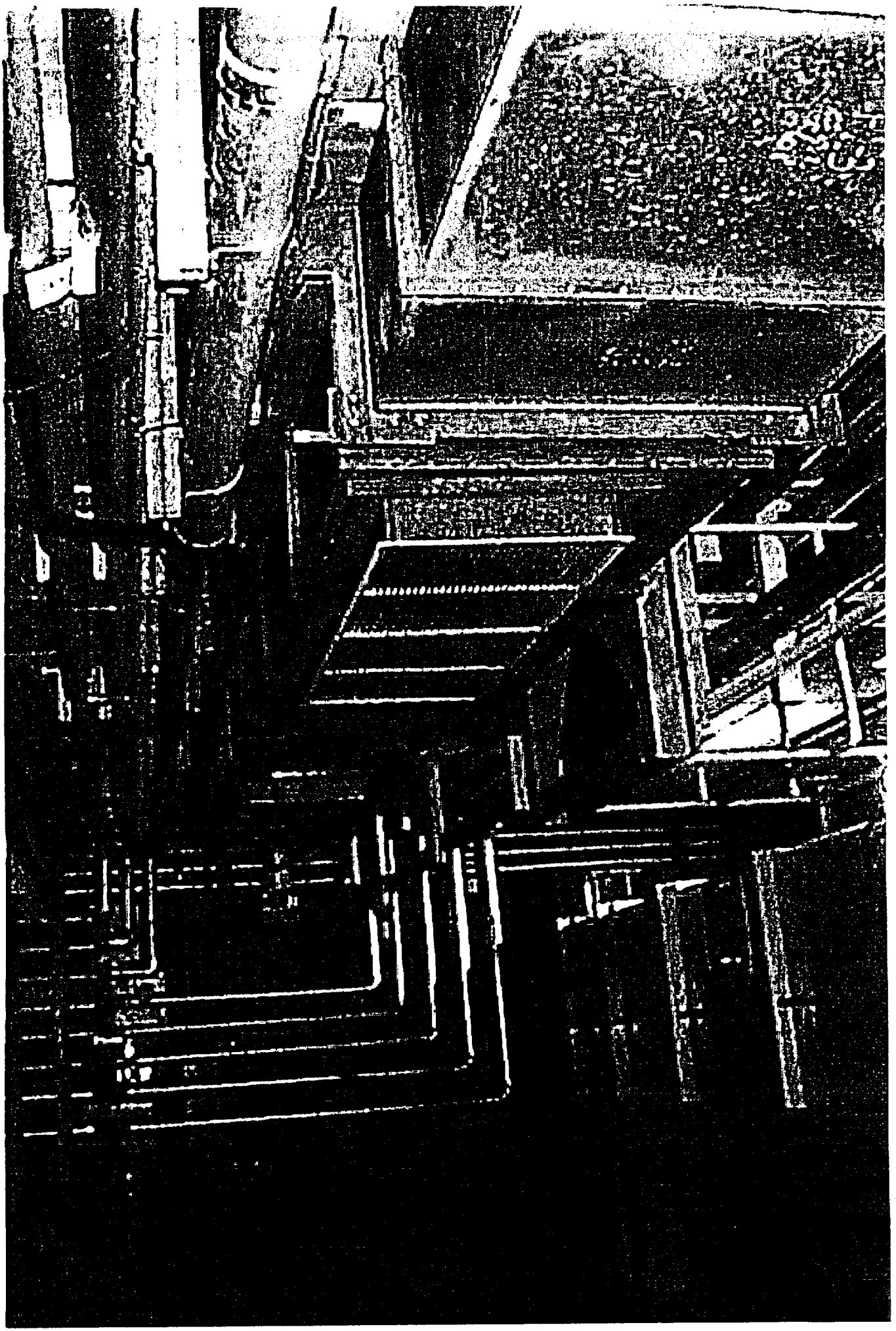

EXHIBIT 16

McGuire Units 1 and 2: March 2, 2000 (LER 369/00/03)(March 30, 2000)

Duke Energy

H. B. Barron
Vice President

Duke Energy Corporation
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-9340
(704) 875-4800 OFFICE
(704) 875-4809 FAX.

DATE: March 30, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station, Unit 1 and 2
Docket No. 50-369
Licensee Event Report 369/00-03, Revision 0
Problem Investigation Process No.: PIP M-00-0844

Gentlemen:

Attached is a Licensee Event Report describing a pre-existing design condition associated with criticality calculations. The condition affects calculations used to generate Limiting Conditions for Operation (LCO) for fuel storage requirements in the spent fuel pool. This event is being reported pursuant to 10 CFR 50.73 (a) (2) (ii) (B) "Operation Outside Design Basis of the Plant". This was previously reported under the parallel criteria of 10 CFR 50.72 in Event Number 36748 on March 2, 2000.

The design basis criteria at issue in this report is the required Keff associated with a spent fuel pool filled with water at zero boric acid concentration. The actual boron acid concentration of the spent fuel pools is maintained in excess of 2500 ppm and monitored on a routine basis as required by technical specifications. These factors mitigate this event to the extent that the condition did not adversely impact plant safety. These actual conditions allow for adequate time to detect and mitigate any dilution of the fuel pool before violating the Keff design basis acceptance criteria.

A Regulatory Commitment is listed as a planned corrective action.

Very truly yours,

H. B. Barron, Jr.
McGuire Nuclear Station, Vice President
Duke Energy Corporation

IE22

Attachment

cc: L. A. Reyes
U.S. Nuclear Regulatory Commission
Region II

Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30323

F. Rinaldi
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

INPO Records Center
700 Galleria Parkway
Atlanta, GA 30339
(Sent Electronically)

S. Shaeffer
NRC Resident Inspector
McGuire Nuclear Station

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 1	05000 369	1 of 5

TITLE (4) Non Conservatism In Spent Fuel Pool Criticality Calculation

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
03	02	00	00	-	03	-	0	03	30	00	McGuire Nuclear Station, Unit 2	05000 370

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (Check one or more of the following) (11)							
		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)	
POWER LEVEL (10)		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)	
		20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)	
		20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)			
		20.405(a)(1)(iv)	XX	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)			
		20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)			

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
M. T. Cash	(704)	875-4117

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS*

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

Unit Status: Both Unit 1 and Unit 2 were in Mode 1 (Power Operation) at 100 percent power at the time of discovery.

Event Description: Modeling methods used to perform spent fuel pool criticality analysis have been determined to be non-conservative. Specifically, certain assumptions may result in Keff in excess of 0.95 for postulated off-normal conditions with 0 ppm boron concentration in the fuel pool. The design basis of the plant requires that fuel stored in the fuel pool remain ≤ 0.95 Keff when fully flooded with unborated water.

Event Cause: This event is the result of an original design condition.

Corrective Action: Technical Specifications will be revised to include additional conservatism to account for uncertainties associated with modeling assumptions.

NRC FORM 366 *NPRDS no longer exists, equipment failures will be reported through EPIX

NRC FORM 366A (6-98) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 6/30/99
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
McGuire Nuclear Station,	05000 369	2000	03	0	2 OF 5

BACKGROUND:

Each unit has an independent fuel storage pool that contains fuel storage racks [EIIS: RK] in a 2 region design. Region 1 uses a high density flux trap design for storage of nuclear fuel. Region 2 uses a high density "egg-crate" design for storage of nuclear fuel. The spent fuel pool storage racks provide for safe storage of nuclear fuel assemblies. This includes maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loading. The rack design provides for fuel storage in a array such that the Neutron Multiplication Factor (Keff) will remain equal to or less than 0.95 assuming unborated water filled the pool. Keff values less than 1.0 indicates a sub-critical condition.

The water in the spent fuel pool contains boric acid dissolved in solution to act as a neutron absorber. The large neutron absorption characteristics of boron in combination with the rack design results in an actual Keff far below 0.95. Technical Specification (TS) 3.7.14, Spent Fuel Pool Boron Concentration, requires that the spent fuel pool boron concentration be within the limits specified in the Core Operating Limits Report (COLR). Current COLR limits require boron concentration > 2675 ppm. TS Surveillance 3.7.14.1, Spent Fuel Pool Boron Concentration Surveillance, requires fuel pool boron verification every 7 days.

TS 3.7.15, Spent Fuel Assembly Storage, also specify acceptable storage configurations for fuel assemblies in the fuel pool. These limits are indexed against the initial enrichment and burnup of individual fuel assemblies. Based on these parameters fuel assemblies are grouped into one of three classes, Filler Assemblies, Unrestricted Storage, and Restricted Storage. This same TS specifies patterns for locating the fuel assemblies based on class. The classification of fuel assemblies and the associated patterns have been determined using nuclear physics models. These models consist of sophisticated neutronic computer codes. The computer codes simulate the geometry, materials, and physical behavior of the nuclear fuel and surrounding materials in the fuel pool. These models have included an assumption that fuel assembly axial burnup distribution is uniform and that axial neutron leakage will be zero. These assumptions along with geometric models have approximated fuel pools as two dimensional systems. The underlying assumption has been that the conservative assumption of zero axial neutron leakage would result in conservative values of Keff. These models have not taken any credit for soluble boron in the spent fuel pools or for other poisons in the form of fuel assembly inserts. The models have taken credit for the boraflex panels [EIIS: PL] in the region 1 racks.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION(6-89)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
McGuire Nuclear Station,	05000 369	2000	03	0	3 OF 5

EVALUATION:

Description of Event

On March 2, 2000, Nuclear Fuel Group engineers in Duke Energy's Corporate Office notified station personnel of a potential non-conservatism in the criticality calculations for the fuel pool storage configurations. Both Unit 1 and Unit 2 were in Mode 1 (Power Operation) at 100 percent power at the time of this notification. Fuel movement was not underway in either units fuel pools at the time of the discovery.

The Nuclear Fuels Group had been performing fuel pool criticality calculations using new models that used 3-dimensional geometry and non uniform fuel assembly axial burnup distributions. These calculations were being performed in support of a proposed TS amendment associated with Boraflex degradation in the spent fuel pools. Results from these analyses caused the Nuclear Fuels Group to suspect previous assumptions regarding the conservatism of 2-dimensional calculations. In the past, it was thought that the range of burnups and enrichments where 2-dimensional calculations were conservative easily bounded fuel assemblies in spent fuel pools. The 3-dimensional calculations estimated that 2-dimensional calculations might become non-conservative at lower burnups and enrichments.

The range at which these non-conservatisms could exist includes burnups and enrichments used to generate the TS limits discussed in the text above. Given the actual fuel assembly burnups and the existing limits, the potential existed that Keff would exceed 0.95 under the postulated unborated condition.

Conclusion

This event did not result in any uncontrolled releases of radioactive material, personnel injuries, or radiation overexposures. This event is not Equipment Performance Information Exchange (EPIX) reportable.

This event is the result of an original design condition.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION(6-99)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/01

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
McGuire Nuclear Station,	05000 369	2000	03	0	4 OF 5

CORRECTIVE ACTION:

Immediate

Verified that the fuel pools were operable with credit for soluble boron concentration maintained at concentrations as required by TS.

Subsequent

An Operating Experience Release was issued for industry awareness of this issue.

Planned

1. Technical Specification limits will be revised to include additional conservatism to account for uncertainties in the 2-dimensional calculations when compared to the 3-dimensional calculations.

2. Upon NRC approval of the TS revision, the Updated Final Safety Analysis Report will be revised to specify storage requirements using Boron credit methodology.

SAFETY ANALYSIS:

Based on this analysis, this event is not considered to be significant. At no time were the safety or health of the public or plant personnel affected as a result of the event.

The design of the spent fuel storage racks assumes the use of unborated water, which maintains each region in a subcritical condition during normal operation with the spent fuel pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 1 or Region 2. This could potentially increase the reactivity of the spent fuel pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the two region poison fuel storage rack with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO.

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station,	05000 369	2000	03	0	5 OF 5

Criticality analysis of the McGuire spent fuel pools demonstrate that approximately 460 ppm of boron for Region 1 and 550 ppm for Region 2 are required to off-set the axial burnup profile uncertainty. This uncertainty was identified as being non-conservative when the 2-dimensional calculation was compared to the 3-dimensional calculation. A boron dilution evaluation for McGuire has documented that for any credible dilution event the minimum soluble boron level in the spent fuel pools would be greater than 937 ppm. This dilution event is based on a minimum boron concentration of 2475 ppm as the initiating point for the event. The results also show that the dilution process requires many hours to significantly reduce pool boron concentration even under the most limiting conditions and provides sufficient time for operator actions to terminate the event. Because of level alarms [EIIS: LA] and operator rounds it is not credible for a dilution of the fuel pool to go undetected for a significant period of time.

Therefore, under conservative assumptions, the fuel pool would be diluted to a boron concentration approximately 400 ppm greater than that needed to maintain the fuel pool below 0.95 Keff. A condition of 0.95 Keff is approximately 5000 pcm subcritical. This is a substantial subcritical margin worth approximately 600 ppm boron concentration assuming a differential boron worth of 8.33 pcm per PPM. As such there is no credible scenario which could have resulted in an inadvertent criticality in the fuel pool under normal or off normal conditions. There are no safety consequences of this event beyond the potential for an inadvertent criticality.

In addition, there have not been any improper loadings of fuel assemblies in the fuel pool in recent operating history that would require consideration of a simultaneous misloading and boron dilution event. This condition had no adverse impact on public health and safety.

EXHIBIT 17

Millstone Unit 2: February 14, 1992 (LER 336/92-003-01)(June 25, 1992)

NRC Form 366 (5-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	1 OF 0 4

TITLE (4)
Spent Fuel Pool Criticality Analysis Error

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	
0 2	1 4	9 2	9 2	- 0 0 3	- 0 1	0 6	2 5	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 3 0	20.402(b)	20.402(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73.(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Robert A. Borchert, Unit 2 Reactor Engineer, Ext. 4418	AREA CODE 2 0 3 4 4 7 - 1 7 9 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	D B	R K	C 4 9 0	N					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 14, 1992, at 1415 hours, with the plant in Mode 1 at 30% power, Northeast Nuclear Energy Company (NNECO) was notified by ABB-Combustion Engineering (ABB-CE) that a calculational error existed in the criticality analysis for the Region 1 spent fuel storage racks. NNECO determined that this condition was reportable as a condition outside of the design basis of the plant. An immediate report was made to the NRC, and the existing reactivity condition of the spent fuel pool was verified to be in compliance with the plant Technical Specifications.

The original effective multiplication factor (K_{eff}) calculated by ABB-CE for the Region 1 fuel storage racks for nominal dimensions, nominal spent fuel pool temperature and 4.5 weight percent enriched fuel assemblies was 0.9224 (without uncertainties). The discovered error results in an underprediction of approximately 0.04 delta K_{eff}. Revised calculations by ABB-CE indicate that K_{eff} is actually 0.963 for the same conditions. An investigation by ABB-CE has traced the error to two approximations used in their calculation.

Criticality analyses to support spent fuel storage rack design changes are complete, and proposed changes to the plant Technical Specifications were submitted to the NRC on April 16, 1992. These changes were approved by the NRC on June 4, 1992.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	9 2	— 0 0 3	— 0 1	0 2 OF 0 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Description of Event

On February 10, 1992, at approximately 1130 hours, Northeast Utilities (NU) was notified by an independent contractor that a higher than expected effective multiplication factor (K_{eff}) was calculated for the Region 1 fuel storage racks. On February 11, 1992, NU notified ABB-Combustion Engineering (ABB-CE) of the potential error in the spent fuel pool criticality analysis. On February 14, 1992, at 1415 hours, with the plant in Mode 1 at 30% power, Northeast Nuclear Energy Company (NNECO) was notified by ABB-CE that a calculational error existed in the criticality analysis for the Region 1 spent fuel storage racks.

The Millstone 2 spent fuel storage racks were modified in May 1986, and consist of two regions:

(a) Region 1 is designed to store up to 384 fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235. Region 1 was designed to allow fuel assembly storage in every location. The Region 1 storage racks contain a neutron poison material (Boroflex), and have a nominal center-to-center pitch of 9.8 inches.

(b) Region 2 is designed to store up to 728 fuel assemblies which have sustained at least 85% of their design burnup. Fuel assemblies are stored in a three-out-of-four array, with blocking devices installed to prevent inadvertent placement of a fuel assembly in the fourth location. The Region 2 storage racks have a nominal center-to-center pitch of 9 inches.

The original effective multiplication factor (K_{eff}) calculated by ABB-CE for the Region 1 fuel storage racks for nominal dimensions, nominal spent fuel pool temperature and 4.5 w/o enriched fuel assemblies is 0.9224 (without uncertainties). The discovered error results in an underprediction of approximately 0.04 delta K_{eff}. Revised calculations by ABB-CE indicate that K_{eff} is actually 0.963 for the same conditions. Evaluations by ABB-CE have confirmed that the Region 2 fuel storage racks are not affected by the error.

NNECO determined that this condition was reportable as a condition outside of the design basis of the plant. An immediate report was made to the NRC, and the existing reactivity condition of the spent fuel pool was verified to be in compliance with the plant Technical Specifications. All fuel movement in the spent fuel pool had previously been restricted due to the observed degradation of the neutron poison material in the Region 1 fuel storage racks. No automatic or manual safety systems were required to respond to this event.

II. Cause of Event

An investigation by ABB-CE has traced the error to two approximations used in their calculation.

First, ABB-CE used an incorrect treatment of the self-shielding effect in Boraflex for the epithermal energy group. This resulted in an overestimation of the neutron absorption in Region 1 and thus a lower calculated K_{eff}.

Second, ABB-CE used a geometric buckling term corresponding to a sparsely populated and unpoisoned array as an approximation of buckling in the poisoned configuration. This approximation also contributed to a lower calculated K_{eff} in Region 1.

III. Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(ii)(B), which requires the reporting of any event or condition that results in the nuclear power plant being in a condition outside the design basis of the plant.

The safety consequence of this event is a potential uncontrolled criticality event in the spent fuel pool. Upon consideration of the following factors, a significant margin to a critical condition was always maintained and, therefore, the safety consequences of this event were minimal:

(a) The boron concentration of the spent fuel pool is procedurally controlled at greater than 1720 ppm, and is typically maintained at greater than 2000 ppm.

(b) All new fuel assemblies previously stored in the Region 1 fuel storage racks had been arranged in a 2 out of 4 checkerboard array.

(c) The maximum initial enrichment of any fuel assemblies previously stored in the Region 1 fuel storage racks was less than 4 weight percent U-235, which is less than the design enrichment of 4.5 weight percent U-235.

(d) All discharged fuel assemblies previously stored in the Region 1 fuel storage racks have sustained at least one cycle of burnup.

IV. Corrective Action

Criticality analyses to support spent fuel storage rack design changes are complete, and proposed changes to the plant Technical Specifications were submitted to the NRC on April 16, 1992. These changes were approved by the NRC on June 4, 1992. These changes split Region 1 into 2 regions, Region A and Region B. Region A can store up to 224 fuel assemblies, which will be qualified for storage by verification of adequate average assembly burnup versus fuel assembly initial enrichment (reactivity credit for burnup). Region B can store up to 120 fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235 and other assemblies which do not satisfy the burnup versus initial enrichment requirements of either Region A or Region C (formerly Region 2). Fuel assemblies will be stored in a 3 out of 4 array in Region B, with blocking devices installed to prevent inadvertent placement or storage of a fuel assembly in the fourth location. Region C is the new designation for the existing Region 2 storage racks. This alphabetic storage rack designation is a human factors consideration, designed to minimize the probability of a fuel assembly movement error and to provide a historical distinction between the various fuel pool configuration records. The attached figure shows the new arrangement of the spent fuel pool.

V. Additional Information

There were no failed components during this event.

Similar LERs: 77-23, 80-05, 83-07, 85-01, 86-10 and 91-10

Spent Fuel Storage Racks

Manufacturer: Combustion Engineering

Model: Hi-Cap Spent Fuel Storage Module

EIIS Code: DB-RK-C490

NRC Form 366A
(5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

Estimated burden per response to comply with this mandatory information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (P-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	9 2	— 0 0 3	— 0 1	0 4 OF 1 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

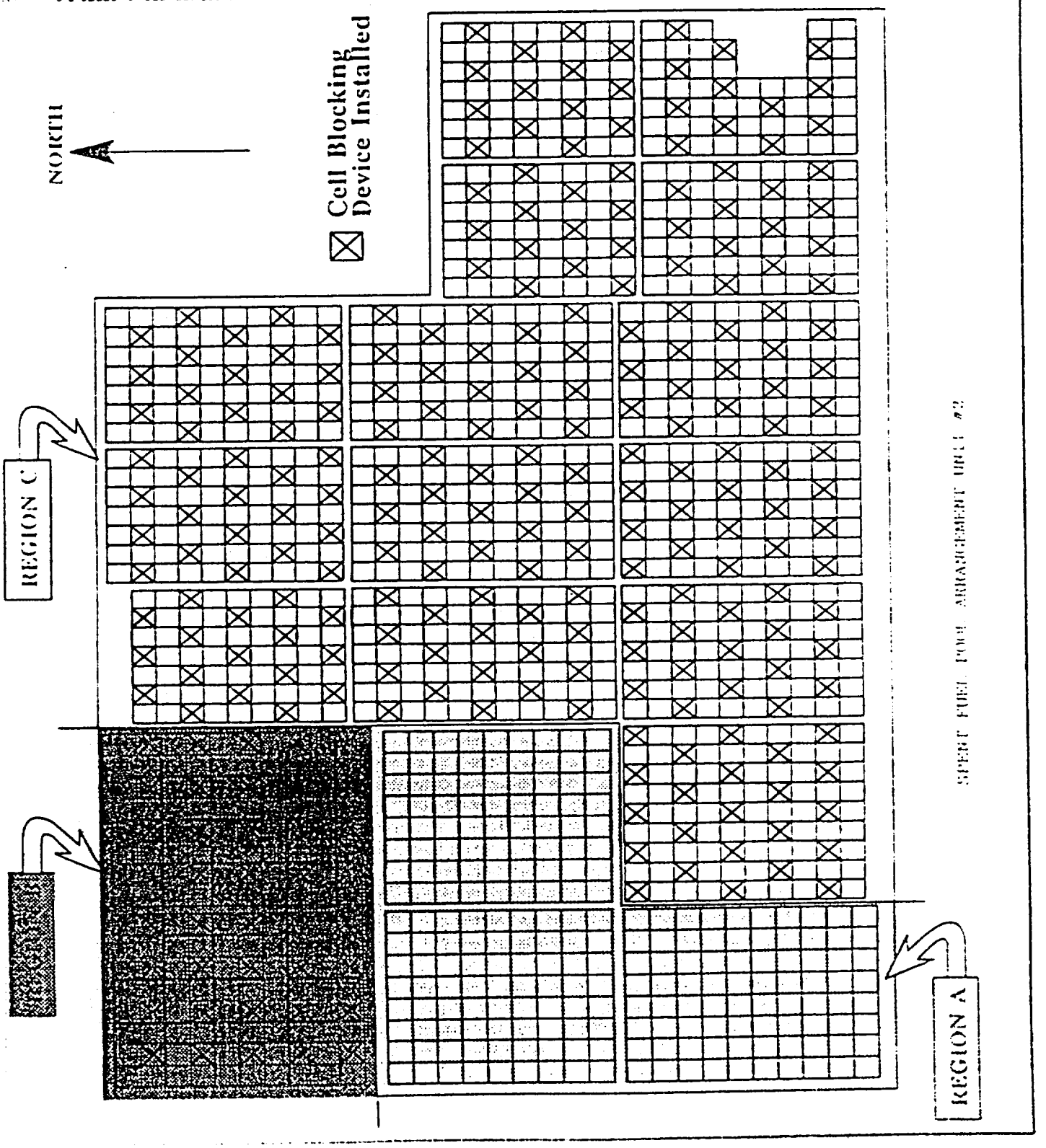

EXHIBIT 18

Millstone Unit 2: (NRC Information Notice 92-21, Supplement 1, Spent Fuel Pool Reactivity Calculations)(April 22, 1992)

EXHIBIT 19

Byron Station: May 28, 1996 (LER 454/96-008-00)(June 25, 1996)

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
BYRON NUCLEAR POWER STATION	05000454	1 OF 9

TITLE (4)

Fuel Assemblies Located In Incorrect Region of Spent Fuel Pool

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
05	28	96	96	-- 008	- 00	06	25	96	FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

CENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David D. Goff, System Engineer X2154	815-234-5441

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 28 May, 1996, Byron Station nuclear engineers confirmed that fuel assemblies F37E, F44E, and G67F were residing in Region 2 of the Spent Fuel Pool (SFP) without meeting the requirements of Technical Specification (TS) 5.6.1.1.b.2, "Fuel Storage - Region 2." The assemblies did not meet the minimum burnup requirements, nor were they checkerboarded. The required minimum burnups were 32651 MWd/MTU, 32651 MWd/MTU, and 32771 MWd/MTU respectively. The actual burnups were 32648 MWd/MTU, 32638 MWd/MTU, and 32728 MWd/MTU respectively.

The cause of this event was cognitive personnel error. The computer spreadsheet used to verify minimum required burnup contained erroneous information for assemblies F37E, F44E, and G67F, and the data in the spreadsheet had not been independently verified. Personnel approving placement of G67F into SFP Region 2 did not have the current revision of Burnup criteria for determination of fuel assembly eligibility for placement into Region 2. Ultimately, the fuel assemblies' burnups were not verified to meet the requirements of TS 5.6.1.1 Amendment 68, "Fuel Storage - Criticality," prior to its implementation.

On 29 May, 1996, the three fuel assemblies were moved into Region 1, as allowed by TS 5.6.1.1.a.2, "Fuel Storage - Region 1." All fuel assemblies remaining in Region 2 were verified either to meet the minimum required burnup or to be stored in a checkerboard pattern.

This event resulted in no safety concerns. The event was bounded by both the older and the newer criticality analyses for Region 2 fuel storage. Adequate reactivity controls were in place to ensure that the k_{∞} limit of 0.95 required by TS 5.6.1.1, "Fuel Storage - Criticality" was not challenged during this event.

This event is reportable under 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's TS.

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 05-28-98 / 1700

Unit 1 Mode 5 - Cold Shutdown . Rx Power Shutdown RCS [AB] Temperature/Pressure 84°F / 0 psig

Unit 1 Mode 4 - Hot Shutdown Rx Power Shutdown RCS [AB] Temperature/Pressure 335°F / 321 psig

B. DESCRIPTION OF EVENT:

Byron Administrative Procedure (BAP) 2000-3T1, "Spent Fuel Burnup Verification Checklist," is a checklist used to verify that fuel assemblies either have or have not accrued the minimum required burnup for uncheckerboarded SFP Region 2 storage. The minimum required burnup is calculated by linear interpolation between values given in BAP 2000-3A1, "Minimum Required Burnup as a Function of Enrichment for Region II High Density Spent Fuel Storage Racks." The values in BAP 2000-3A1 are intended to bound TS Figure 5.6-1, "Minimum Burnup Versus Initial Enrichment For Region 2 Storage."

On 10 February, 1993, Byron Station nuclear engineers (engineers 1 and 2) completed BAP 2000-3T1 for fuel assemblies including F37E and F44E. The checklist showed both assemblies with an initial enrichment of 3.8 wt% U-235 and a minimum required burnup for placement into Region 2 of 32540 MWd/MTU, given by BAP 2000-3A1 Rev 1. F37E and F44E had accrued actual burnups of 32648 MWd/MTU and 32638 MWd/MTU respectively. The minimum value of 32540 MWd/MTU was appropriate for an initial enrichment of 3.8 wt% U-235, and both assemblies met the Technical Specification requirement for uncheckerboarded Region 2 storage.

On 11 February, 1993, Nuclear Fuels Services (NFS) issued letter NFS:PSS:93-060 which, in part, stated that fuel assemblies F37E and F44E met the minimum burnup requirements of TS 5.6.1.1. This letter showed F37E and F44E having accumulated 32648.0 MWd/MTU and 32638.4 MWd/MTU respectively.

On 18 August, 1993, Byron Station fuel handlers moved fuel assemblies F37E and F44E into SFP locations K-C2 and K-D8, respectively, in Region 2. The assemblies were not stored in a checkerboard pattern since they met the minimum required burnup restrictions presently in place. The moves were performed in accordance with page 93-104 of an approved BAP 2000-3T3 Rev 1, "PWR Station Nuclear Component Transfer List." Engineers 1 and 3 verified that BAP 2000-3T1 was completed prior to transfer list approval.

Starting in the summer months of 1994, engineer 3 was assisting in the preparation of a license amendment request. This request would allow storage of fuel in Region 2 up to 5.0 wt% U-235 and was supported by a new criticality analysis.

On 11 August, 1994, Byron Station engineers (engineers 3 and 4) initiated Problem Identification Form (PIF) 454-201-94-69200. This PIF documented that Byron Station and NFS employed different methods in determining whether a fuel assembly meets the minimum burnup requirement for Region 2 storage. NFS used a polynomial fit through the points given in the criticality analysis after applying a 1.03 multiplicative penalty to account for fit error and uncertainty in the assembly burnup calculation. Byron Station used linear interpolation between points which bound TS Figure 5.6-1 Amendment 25. This PIF also identified that TS Figure 5.6-1 Amendment 25 did not, for all initial enrichments, bound the criticality analysis used as the basis for the curve.

B. DESCRIPTION OF EVENT (cont.)

Byron Station and NFS continued to use different criteria for minimum required burnup determination. The license amendment request being developed, when approved, would render the second problem moot. For the interim, engineer 3 prepared a revision request for BAP 2000-3A1 to change the points used for minimum burnup determination such that both TS Figure 5.6-1 Amendment 25 and the criticality analysis would be bounded.

On 16 September, 1994, Byron Station nuclear engineers (engineers 5 and 6) completed BAP 2000-3T1 for fuel assemblies including G67F. This checklist showed the G67F assembly with an initial enrichment of 3.809 wt% U-235 and meeting the minimum required burnup for placement into Region 2 of 32661 MWd/MTU. G67F had accrued an actual burnup of 32728 MWd/MTU. The minimum value of 32661 MWd/MTU was conservative for an initial enrichment of 3.809 wt% U-235. Engineer 6 stated that the enrichment value was conservatively rounded up to 3.81 wt% U-235 when the minimum required burnup was calculated. G67F met the Technical Specification requirement for uncheckerboarded Region 2 storage.

Also on 16 September, 1994, NFS issued letter NFS:PSS:94-225 which, in part, stated that fuel assembly G67F did not meet the minimum burnup requirements of TS 5.6.1.1. The discrepancy between the Byron Station and NFS conclusions resulted from the different methods in determining eligibility of a Region 2 storage candidate. Since G67F had accrued the minimum required burnup in accordance with BAP 2000-3A1 Rev 1, it was deemed to be suitable for uncheckerboarded Region 2 storage.

On 20 October, 1994, Byron Station Onsite Review (OSR) 94-078 approved a license amendment request for Byron Station Units 1 and 2 Technical Specifications. This amendment request later became TS Amendment 68. This request would, in part, revise Figure 5.6-1 Amendment 25 to be conservativ 3% greater than the new criticality analysis. Discrete values would be provided in Figure 5.6-1 along with ..structions that would allow linear interpolation between the values. In particular, the required burnup for an initial enrichment of 3.8 wt% U-235 would be increased from 32640 MWd/MTU to 32651 MWd/MTU.

The OSR 94-078 package did not document the review of incumbent fuel assemblies and their eligibility for Region 2 storage with the new minimum burnup curve. Engineer 3 and a representative from NFS participated in the OSR.

However, Byron Station nuclear engineers (engineers 3 and 7) had conducted a review of the incumbent fuel assemblies over the course of several months from approximately August to November, 1994. This review was performed by engineer 7 building a computer spreadsheet to calculate assembly eligibility, and then the ouput was spot checked by engineer 3 for verification. The spreadsheet required input data for initial enrichment, storage location, and actual accrued burnup, and then checked each fuel assembly against several minimum burnup criteria, including those that would become BAP 2000-3A1 Rev 2 and TS Amendment 68. The spreadsheet calculation produced a Boolean output for each assembly, i.e., "OK" or "not OK" for uncheckerboarded Region 2 storage.

Initial enrichment, storage location, and actual accrued burnup data loaded into the spreadsheet for F37E, F44E, and G67F were incorrect. This resulted in the spreadsheet producing erroneous "OK" outputs for those assemblies. Had correct data been loaded into the spreadsheet, the assemblies would have been properly identified as "not OK" when compared against the minimum required burnups of BAP 2000-3A1 and TS Amendment 68.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
BYRON NUCLEAR POWER STATION	05000454	95	- 008	- 00	4 OF 9

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

B. DESCRIPTION OF EVENT (cont.)

On 26 October, 1994, PIF 454-201-94-69200 was closed with the understanding that Byron Station and NFS would continue to use different methods for determining minimum required burnup for Region 2 storage. This would serve as a diverse means to identify assemblies suitable for Region 2 storage.

On 13 December, 1994, Byron Station OSR approved revision 2 of BAP 2000-3A1. This revision was processed as a corrective action to PIF 454-201-94-69200, which identified that TS Figure 5.6-1 Amendment 25 did not, for all initial enrichments, bound the criticality analysis used as the basis for the curve. The new revisic bounded both the criticality analysis and TS Figure 5.6-1 Amendment 25. Under the new revision, the minimum required burnup for an initial enrichment of 3.8 wt% U-235 was increased from 32540 MWd/MTU to 32800 MWd/MTU. Byron Station took credit for the review performed in association with OSR 94-078 to verify compliance of the incumbent fuel assemblies. As stated before, the spreadsheet contained erroneous data for F37E, F44E, and G67F. Hence, all three assemblies passed the review. Under BAP 2000-3A1 Rev 2, fuel assemblies F37E, F44E, and G67F no longer met the minimum required burnup, though they all met the requirements of revision 1.

On 20 January, 1995, the Nuclear Regulatory Commission (NRC) issued Amendment 68 to Byron Station Units 1 and 2 TS, revising Figure 5.6-1 as requested under the licensing amendment request previously submitted.

On 23 January, 1995, Byron Station fuel handlers moved fuel assembly G67F into SFP location G-L12 in Region 2. The assembly was not stored in a checkerboard pattern since it had been verified to meet the requirements of BAP 2000-3A1 Rev 1. This was done in accordance with page 95-5 of an approved PWR Station Nuclear Component Transfer List. Engineers 5 and 8 verified that BAP 2000-3T1 Rev. 1 was completed prior to transfer list approval. However, BAP 2000-3T1 Rev. 1 had been completed in September, 1994, using BAP 2000-3A1 Rev 1. BAP 2000-3A1 Rev. 2 was now the current revision, and assembly burnups should have been compared to revision 2 requirements rather than the revision 1 requirements. The assembly did not meet the minimum burnup requirement of BAP 2000-3A1 Rev 2 or TS Amendment 68, though it did comply with TS Figure 5.6-1 Amendment 25.

On 25 January, 1995, Byron Station OSR 95-007 approved for use Amendment 68 and its implementation plan. The OSR 95-007 package acknowledged that TS Figure 5.6-1 was changing. The implementation plan stated that the Byron Station nuclear engineering group "will revise BAP 2000-3A1 to reflect the new burnup curve to identify assemblies that are acceptable to load in Region 2." At that time, it was thought that BAP 2000-3A1 Rev 2 was more conservative than TS Figure 5.6-1 Amendment 68. Therefore, the implementation plan required no deadline for revision of BAP 2000-3A1. The OSR package did not discuss the review that had been performed of the incumbent assemblies. Engineer 5 and the Station Reactor Engineer (SREI participated in the OSR.

On 30 January, 1995, Byron Station OSR approved revision 3 of BAP 2000-3T2, "NCTL Verification Checklist." This revision provided more explicitly detailed guidance on how to perform the verification of minimum required burnups on BAP 2000-3T1.

On 6 February, 1995, Byron Station OSR approved revision 2 of BAP 2000-3T1. This revision added more documentation of information so that minimum required burnups could be more readily and accurately determined.

B. DESCRIPTION OF EVENT (cont.)

On 1 March, 1995, all TS manual holders were instructed, in a letter from the Byron Station Regulatory Assurance Department Supervisor, to implement TS Amendments 67, 68, and 69. At this time, assemblies F37E, F44E, and G67F, were in Region 2 and were in violation of TS 5.6.1.1. Each had been previously approved for residence in Region 2 using a revision of BAP 2000-3A1 which reflected an earlier TS amendment.

On 17 August, 1995, Byron Station OSR approved revision 3 of BAP 2000-3A1. This revision was processed due to TS Amendment 68 changing the minimum required burnup curve. The procedure now exactly matched TS Figure 5.6-1, requiring 32651 MWd/MTU for an initial enrichment of 3.8 wt% U-235. Again, Byron Station took credit for the review performed in association with OSR 94-078 to verify compliance of the incumbent fuel assemblies. Two fuel assemblies were moved into SFP Region 2 since implementation of TS Amendment 68 on 1 March, 1995. They were moved from failed fuel canisters on 1 June and 29 June. Both assemblies met the minimum burnup requirement.

On 24 May, 1996, while performing BAP 2000-3T1 for fuel assemblies anticipated to be moved in association with upcoming spent fuel storage rack neutron attenuation testing, Byron Station nuclear engineers (engineers 7 and 9) found indications that fuel assemblies F37E and F44E did not meet the minimum burnup as required by TS 5.6.1.1.b.2.a, "Fuel Storage - Region 2." Nor were these two assemblies stored in a checkerboard pattern as allowed by TS 5.6.1.1.b.2.b, "Fuel Storage - Region 2." Byron Station contacted NFS for verification of actual burnup and minimum required burnup and to assist the investigation into whether these fuel assemblies were incorrectly residing in Region 2.

On 28 May, 1996, while performing BAP 2000-3T1 for fuel assemblies anticipated to be moved in association with upcoming spent fuel storage rack neutron attenuation testing, Byron Station nuclear engineers (engineers 7 and 9) found indications that fuel assembly G67F did not meet the minimum burnup as required by TS 5.6.1.1.b.2.a. Nor was this assembly stored in a checkerboard pattern as allowed by TS 5.6.1.1.b.2.b. Byron Station again contacted NFS for verification of actual burnup and minimum required burnup and to include this fuel assembly in the investigation.

On 28 May, Byron Station nuclear engineers (engineers 7, 9 and the acting SRE) and NFS held a conference call discussing the results of the NFS investigation into fuel assemblies F37E, F44E, and G67F. It was determined at 17:00 that all three assemblies were in violation of TS 5.6.1.1.b.2.

C. CAUSE OF EVENT:

The cause of F37E and F44E being incorrectly stored in Region 2 was cognitive personnel error. The data used by the computer spreadsheet for verifying minimum required burnup was not entered correctly nor was it independently verified to be accurate. The spreadsheet data failed to show that F37E and F44E were in SFP Region 2. Furthermore, the spreadsheet data failed to use the correct burnup values for F37E and F44E. This resulted in assemblies F37E and F44E producing erroneous "OK" spreadsheet outputs. This faulty technical review was part of the basis for the Byron Station OSR 95-008 approval and acceptance of TS Amendment 68. The amendment was then implemented with plant conditions not conforming to the new requirements.

C. CAUSE OF EVENT (cont.)

The cause of G67F being incorrectly stored in Region 2 was also cognitive personnel error. Personnel approving the NCTL to place G67F in SFP Region 2 failed to use the current procedure revision of BAP 2000-3A1 to verify that G67F had accrued the minimum required burnup for uncheckerboarded Region 2 storage. The previous revision that was used did not reflect current plant conditions. This resulted in an ineligible fuel assembly being placed into Region 2.

D. SAFETY ANALYSIS:

The SFP condition throughout this event was bounded by the two criticality analyses used as the bases for TS Figure 5.6-1 prior to and after Amendment 68. All uncheckerboarded fuel assemblies, including F37E, F44E, and G67F, met the minimum burnup requirements of those analyses. However, the SFP condition failed to meet the current TS requirement, which was 3% greater than the current criticality analysis.

UFSAR section 9.1.3.2 addresses the safety evaluation for storing spent fuel in the SFP. The criticality portion is based on the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Considering Boraflex Gaps and Shrinkage" document from Westinghouse dated June, 1994, as amended by 94CB*-G-0105 and 94CB*-G-0142. Section 5.0, Discussion of Postulated Accidents, addresses an abnormal condition where reactivity would increase beyond the analyzed condition: a fuel assembly is misloaded into Region 2 which does not satisfy the requirements.

While, in the scenario considered, only one assembly is misloaded, the analysis makes several conservative assumptions:

1. All fuel assemblies contain U-235 at the nominal enrichment or its equivalent at the minimum required burnup.

2. All fuel assemblies are uniformly enriched. No credit is taken for reduced-enrichment or natural uranium axial blankets.

3. No credit is taken for U-234, U-236, or any fission product poisons. No credit is taken for any burnable absorber material which may remain in the fuel.

4. All storage locations are loaded with fuel assemblies not containing any absorption material.

5. The storage locations are infinite in lateral extent.

6. The array is moderated by pure water of 1.0 g/cc.

7. A conservative Boraflex degradation model is assumed.

8. The scenario where a fresh assembly with an enrichment of 4.2 wt% is inserted into a 5x5 array of the nominal assemblies is considered.

D. Safety Analysis (cont.)

The maximum k_{eff} at a 95% probability with 95% confidence and including the statistical summation of independent uncertainties is 0.9449 for Region 2 under the nominal conditions. The increase in reactivity due to the misloaded assembly is no more than 0.0438 delta k. However, only a single failure must be accounted for, so soluble boron may be credited. The reactivity from 300 ppm boron is approximately -0.06 delta k, more than offsetting the increase from the misloading. Thus, the k_{eff} limit of 0.95 required by TS 5.6.1.1 is not challenged during this abnormal condition.

The situation described in this report, with three fuel assemblies misloaded rather than just one, is more conservative than the accident analysis due to the following considerations:

1. Nearly all fuel assemblies residing in Region 2 exceed the minimum burnup requirement, making them less reactive than the reference assemblies.

2. Many fuel assemblies have reduced-enrichment or natural uranium axial blankets of six inches at both ends, reducing their reactivities.

3. All fuel assemblies contain U-234 and U-236, and spent assemblies contain fission product poisons as well. These materials further reduce reactivity.

4. Not every storage location contains fuel. Locally, there are several empty locations. Some of the fuel assemblies contain absorber material such as rod cluster control assemblies (RCCAs).

5. The SFP is finite, exhibiting nonzero neutron leakage at the boundaries.

6. The water in the SFP is normally approximately 80 degF, having a density less than 1.0 g/cc. Soluble boron concentration in the SFP remained greater than 1280 ppm since January, 1995, providing at least -0.22 delta k reactivity.

7. Previous neutron attenuation testing results imply that the Boraflex in Region 2 has not deteriorated to the extent assumed in the analysis.

8. The improperly located fuel assemblies are significantly less reactive than the fresh 4.2 wt% enriched assembly assumed in the accident analysis. Fuel assemblies F37E, F44E, and G67F fell short of the required burnup by 3 MWd/MTU, 13 MWd/MTU, and 43 MWd/MTU respectively. These values are within approximately 0.1% of the required burnup values.

The combination of the above factors ensured that the k_{eff} limit of 0.95 required by TS 5.6.1.1 was not challenged during this event.

E. CORRECTIVE ACTIONS:

On 28 May, 1998, at 17:15, Byron Station nuclear engineers initiated PIF 454-180-98-0008, identifying three fuel assemblies inappropriately residing in Region 2 of the SFP. Byron Station Regulatory Assurance, Operations, and System Engineering management were notified. The NRC Resident Inspector was also notified.

Concurrently, NFS initiated PIF 901-201-98-07800 identifying possible inadequacies and inconsistencies in their methods of determining eligibility of Region 2 candidate fuel assemblies. The investigation results show that these inadequacies and inconsistencies did not contribute to the root causes of this event.

On 29 May, 1998, at 05:15, Byron Station fuel handlers moved fuel assemblies F37E, F44E, and G67F into SFP storage locations in Region 1. This was done in accordance with page 98-103 of an approved PWR Station Nuclear Component Transfer List.

NFS subsequently performed a review of all fuel assemblies residing in Region 2 using TS Amendment 68 criteria. This review was transmitted as NFS:PSS:98-142 and PSSCN:98-023. It consisted of a list of every fuel assembly in the Byron Station SFP as of 31 March, 1998, and identified which assemblies had achieved the minimum required burnup for Region 2 storage. Byron Station engineers 7 and 9 then verified that those assemblies not meeting minimum burnup were either stored in Region 1 or in a checkerboard pattern. There were no assemblies stored inappropriately in Region 2. All fuel moves into Region 2 performed since 31 March, 1998, have had eligibility requirements verified in accordance with BAP 2000-3A1 Rev 3.

BAP 2000-3T2 Rev 3 is currently in place and provides explicit guidance on the preparation and independent review of BAP 2000-3T1 Rev. 2. This revision was not in place at the times F37E, F44E, and G67F were approved for uncheckerboarded Region 2 storage. The guidance provided presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

BAP 2000-3T1 Rev. 2 is currently in place and provides improved documentation of minimum required burnup for fuel assemblies being moved to or within Region 2. This revision was not in place at the times F37E, F44E, and G67F were approved for uncheckerboarded Region 2 storage. The improved documentation shows initial enrichment, minimum required burnup, and actual accrued burnup for each assembly and presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

BAP 2000-3A1 Rev. 3 is currently in place and is identical to the requirements of TS Figure 5.6-1 Amendment 68 as well as the current NFS method of determining Region 2 storage eligibility. All future fuel assemblies approved for Region 2 storage will have minimum required burnups determined in accordance with this procedure or its equivalent. Any future TS Amendment changing TS Figure 5.6-1 will have a concurrent revision to BAP 2000-3A1 associated with it reflecting the new requirements. This presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

Performance expectations have been discussed with persons involved in the errors that contributed to this event.

This LER will be discussed with all members of the Byron Station nuclear engineering group, emphasizing personnel performance expectations. A copy will be placed in the nuclear engineering group required reading book. NTS item 454-201-98-0008-01 tracks completion of this action.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

LER 454:94-006, "Fuel Assembly Located in Wrong Region of Spent Fuel Pool due to Personnel Error," documents a similar event. On 15 July, 1994, SED found a fuel assembly in Region 2 that neither met the minimum burnup requirements of TS Figure 5.6-1 nor was checkerboarded. The cause of this event was determined to be cognitive personnel errors. The Nuclear Materials Custodian and an independent reviewer failed to use the approved method to verify assemblies met the minimum burnup requirements for storage in Region 2.

Although the 454:94-006 event resulted in a fuel assembly incorrectly residing in SFP Region 2, the circumstances leading to this event were different from those leading to the 454-180-96-0008 event.

G. COMPONENT FAILURE DATA:

No components failed in association with this event.

EXHIBIT 20

Farley Unit 1: March 23, 2000 (LER 348/2000-004-00)(April 20, 2000)

Dave Morey
Vice President
Farley Project

Southern Nuclear
Operating Company, Inc.
Post Office Box 1295
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SOUTHERN COMPANY
Energy to Serve Your World℠

April 20, 2000

Docket No.: 50-348

NEL-00-0112

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Joseph M. Farley Nuclear Plant
Unit 1 Licensee Event Report 2000-004-00
Three Spent Fuel Assemblies in Spent Fuel Pool
Locations Not Allowed By Technical Specification 3.7.15

Ladies and Gentlemen:

Joseph M. Farley Nuclear Plant Unit 1 Licensee Event Report (LER) No. 2000-004-00 is being submitted in accordance with 50.73(a)(2)(i). There are two NRC commitments in the LER. They are as follows:

1) The applicable procedure will be changed to provide sufficient detail to ensure correct configuration determinations and define independent review requirements prior to moving fuel.

2) Responsible personnel will be trained on lessons learned from this event, review requirements, and revisions to the procedure prior to moving fuel.

These will be completed prior to the next fuel assembly movement.

If you have any questions, please advise.

Respectfully submitted,

DM Morey

Dave Morey

EWC/maf ler200004-00.doc
Attachment

JE22

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. M. Padovan, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

NRC FORM 366 (6-1998) U.S.NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 06/30/2001
Estimated burden per response to comply with this mandatory information request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	0 5 0 0 0 3 4 8	1 OF 4

TITLE (4)
Three Spent Fuel Assemblies in Spent Fuel Pool Locations Not Allowed by Technical Specification 3.7.15

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										0 5 0 0 0
0 3	2 3	2 0 0 0	2 0 0 0	0 0 4	0 0	0 4	2 0	2 0 0 0	FACILITY NAME	0 5 0 0 0

OPERATING MODE (9)	6
POWER LEVEL (10)	0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include area code)
L. M. Stinson, General Manager Nuclear Plant	3 3 4 - 8 9 9 - 5 1 5 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

On March 23, 2000 at 0830, it was determined that Unit 1 had been operated in a condition contrary to Technical Specification (TS) 3.7.15, in that three spent fuel assemblies were loaded in the Spent Fuel Pool in configurations contrary to TS Figures 4.3-1 through 4.3-5. This condition first occurred during the core offload for the current refueling cycle on March 13, 2000 at 1449.

Manual verification of the acceptability of proposed offload configuration on March 11, 2000 failed to identify that three assemblies had insufficient burnup for their planned storage locations. On March 23, 2000, while Reactor Engineering personnel were loading the fuel location data into a Special Nuclear Materials tracking software package being developed for use, three fuel assemblies that did not meet the Technical Specification storage configuration requirements were identified. On March 23, 2000 at 0933, relocation of the three affected assemblies into acceptable locations was completed.

This event was caused by personnel error in that personnel responsible for developing, performing, and verifying the SFP configuration failed to assure that three fuel assemblies met the Technical Specification configuration requirements. Contributing causes were lack of detail in the procedure, experience level of personnel performing this evolution, and insufficient independent review in the verification process. The procedure will be changed to provide sufficient detail to ensure correct configuration determinations. Responsible personnel will be trained on revisions to this procedure and the independent review requirements prior to moving fuel.

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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

Westinghouse – Pressurized Water Reactor
Energy Industry Identification Codes are identified in the text as [XX].

Description of Event

On March 23, 2000 at 0830, it was determined that Unit 1 had been operated in a condition contrary to Technical Specification (TS) 3.7.15, in that three spent fuel assemblies were loaded in configurations contrary to TS Figures 4.3-1 through 4.3-5. This condition first occurred during the core offload for the current refueling cycle on March 13, 2000 at 1449.

On March 10 and 11, 2000, Reactor Engineering personnel reviewed the proposed configuration for the Spent Fuel Pool (SFP) for the Sixteenth Refueling Outage core offload against the TS.

The following combination of circumstances created an error likely situation for performance of this evolution: As the SFP approaches capacity with time, the complexity of the task of determining acceptable storage configurations has increased, however, the procedure had not been strengthened to address this additional complexity. The performance of this evolution was initially started using conservative fuel burnups. This resulted in excessive conservatisms being applied to the determination of acceptable configurations, and the evolution was restarted using actual end of cycle burnups. This reduced the time available for completion of the activity. As a result, personnel performing the verification and review chose to perform the activity together instead of sequentially, resulting in a reduction in quality of the review.

Manual verification of the acceptability of proposed offload configuration failed to identify that the proposed configuration would not meet the acceptable configurations defined in TS Figures 4.3-1 through 4.3-5, for three spent fuel assemblies. The review of this verification process also failed to identify this condition. The assemblies in question had burnups of up to 3300 Megawatt-days per Metric Ton Uranium (MWD/MTU) less than the minimum required for the proposed storage locations. The core offload was performed from March 11 through 14, 2000.

On March 23, 2000, while Reactor Engineering personnel were loading the fuel location data into a Special Nuclear Materials tracking software package being developed for use, these three fuel assemblies that did not meet the acceptable loading patterns were identified. On March 23, 2000 at 0933, relocation of these three affected assemblies into acceptable locations was completed.

NRC FORM 366A (6-1998) U.S.NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

Cause of Event

This event was caused by personnel error in that personnel responsible for developing, performing, and verifying the SFP configuration failed to assure that three fuel assemblies met the Technical Specification configuration requirements. Contributing causes were lack of detail in the procedure, experience level of personnel to perform this evolution, and insufficient independent review in the verification process.

Safety Assessment

Analysis shows that a boron concentration of 700 ppm would have kept Keff below the limit of 0.95. Since the Technical Specifications require a minimum boron concentration in the SFP of 2000 ppm, and actual boron concentration was 2435 ppm, the Keff of the SFP remained less than 0.95 throughout this event. In addition, this analysis conservatively took no credit for the Boraflex neutron adsorber located in the SFP racks

Therefore the health and safety of the public were unaffected by this event.

This event does not represent a Safety System Functional Failure.

Corrective Action

On 3/23/2000 the three assemblies were relocated to acceptable configurations.

The Unit 2 SFP was checked for fuel in incorrect storage configurations. None was identified.

The applicable procedure will be changed to provide sufficient detail to ensure correct configuration determinations and define independent review requirements prior to moving fuel.

Responsible personnel will be trained on lessons learned from this event, review requirements, and revisions to the procedure prior to moving fuel.

NRC FORM 366A (6-1998)

U.S.NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)		
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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

Additional Information

As an enhancement, a computerized SFP configuration verification system will be placed in service prior to September 30, 2000. The configuration verification procedure will be revised to reflect the computerized verification process, and optimize the manual verification process, by September 30, 2000. Reactor Engineering personnel and supervision will be trained on the software additions and related procedure changes by October 30, 2000.

A voluntary 4-hour nonemergency notification was made to the NRC at 1215 on March 23, 2000.

The following LER has been submitted in the past 2 years on a combination of personnel error and inadequate procedure:

LER 1998-003-00 Unit 1, Waste Gas Decay Tank Hydrogen and Oxygen Exceeded Concentration Limits