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SVP-03-118

November 28, 2003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Quad Cities Nuclear Power Station, Unit 1
Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Transmittal of Lost Parts Analysis and Associated Operability Evaluation

On November 12, 2003, Exelon Generation Company, LLC (EGC) commenced a shutdown of Quad Cities Nuclear Power Station (QCNPS), Unit 1, as a result of suspected steam dryer degradation. The degradation was suspected due to noted increases in moisture carryover and corresponding changes in main steam flow. Following reactor vessel disassembly, EGC completed detailed inspections of 100% of the accessible exterior and interior areas of the QCNPS Unit 1 steam dryer. Details of the inspections were discussed with the NRC in a conference call on November 20, 2003.

It was identified during the November 13, 2003, initial inspection that a portion of the damaged steam dryer outer hood bank was missing. The original size of the loose part is approximately 6.5" x 9.0" x 0.5", and it is unknown if the part remained as a single piece or broke into multiple pieces. In a conference call with the NRC on November 25, 2003, EGC discussed the comprehensive inspections that have been performed in an attempt to locate the missing dryer material. To date, the specific location of the missing dryer material has not been identified.

An evaluation addressing safety and operational concerns has been performed by General Electric (GE), and reviewed and accepted by EGC, to support plant operation for the remainder of the current operating cycle without recovery of the loose part. On November 26, 2003, the NRC requested EGC to submit the evaluation for NRC review. Attachment 1 provides the requested information.

Attachment 1 describes the potential migration paths of the loose part and concludes that safe reactor operation will not be compromised with the presence of the lost part(s) in the reactor vessel. Based on inspections of the potential migration paths of the lost part, EGC has concluded that the most probable location for the lost part is in the bottom head region of the reactor vessel. Attachment 1 concludes that a lost part in the bottom head region is not a safety concern, but may be an operational concern that can be adequately managed.

EGC recognizes and is sensitive to the issue of long-term reactor operation without retrieval of the missing dryer material. Therefore, EGC will make a decision prior to the next QCNPS

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Unit 1 refueling outage as to whether to continue the effort to locate and retrieve the missing dryer material. In part, that decision will include assessing the ramifications associated with performing an inspection of the bottom head region. This assessment will consider factors such as dose required to perform the inspection, safety consequences of continued operation without retrieval of the lost part, and the ability to retrieve the part if located. In the interim, an operability evaluation has been performed to support plant operation until the next refueling outage without retrieving the part. Attachment 2 provides the operability evaluation. If the outcome of the decision is to discontinue efforts to locate the missing dryer material, then the appropriate engineering evaluations will be performed for long-term operation, including a review in accordance with the provisions of 10 CFR 50.59, "Changes, tests, and experiments."

Attachment 1 contains GE proprietary information that should be withheld from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the affidavit included with Attachment 1. EGC hereby requests that Attachment 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790, "Public inspections, exemptions, requests for withholding," and 10 CFR 9.17, "Agency records exempt from public disclosure." The basis for the proprietary determination is documented in the affidavit. The NRC is not requested to approve Attachment 1 or to issue any documentation regarding plant operation based on Attachment 1. Consequently, Attachment 1 is submitted "voluntarily" consistent with 10 CFR 2.790(b)(4)(ii), and a non-proprietary version of Attachment 1 is not available.

Should you have any questions concerning this letter, please contact Mr. Wally J. Beck at (309) 227-2800.

Respectfully,

Timothy J. Tulon

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

Attachments:
1. GE-NE-0000-0023-5200-R0, "Lost Parts Analysis for Quad Cities Generating Station Unit 1 Steam Dryer Outer Hood (270° Side)," dated November 2003
2. Operability Evaluation Supporting Unit 1 Operation with Lost Part

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

ATTACHMENT 2

Operability Evaluation Supporting Unit 1 Operation with Lost Part

1.0 ISSUE IDENTIFICATION:

1.1 CR #: 188333

1.2 OpEval #: 188333-08 Revision: 1

General Information:

1.3 Affected Station(s): Quad Cities

1.4 Unit(s): One

1.5 System: RX/0202

1.6 Component(s) Affected: Reactor Vessel and Internals

1.7 Detailed description of what SSC is degraded or the nonconforming condition and by what means and when first discovered:

On 11/12/03, Quad Cities Unit 1 entered the 51st forced shutdown (Q1F51) to inspect the steam dryer for damage based on observed changes in main steamline flows, steamline pressure drop, and increasing moisture carryover measurements. The symptoms observed were consistent with previous events on Unit 2 that had resulted in the discovery of damage to the steam dryer. Subsequent inspections on Unit 1 identified damage to several areas of the dryer. A General Electric Inspection Notification Report (INR) was written for each specific indication identified. The reports are sequentially numbered in the following format: INR-Q1F51-03-XX. INR-Q1F51-03-01, Revision 1 identified damage in the outer hood F bank. This damage was a crack at the top corner portion of the hood that extended horizontally towards the center of the hood and downward into the vertical position of the hood. The crack terminates in the vertical section where a portion of the dryer was missing. This missing part of the steam dryer outer bank hood is approximately 6.5 inches by 9.0 inches (see Attachment A). It is assumed that a lost part(s) the size of this opening or smaller was generated due to fatigue cracking. An extensive search of the lost part(s) was performed as documented in EC 345951 Revision 1. No part(s) of the missing piece were located. However, based upon inspections of the 1B Recirculation Pump, impact marks were identified on the pump's impeller that were indicative of the missing part(s) passing through the pump. The pump impeller was replaced during Q1F49 because of the Jet Pump Beam Failure Event. This condition is documented in CR 188260. CR 188333 was written to address the impact of the lost part(s) of dryer passing through the 1B recirculation pump. Therefore, this operability support documentation evaluates Unit 1 operation with the lost part(s) as a degraded condition until retrieved during Q1R18 or the 50.59 processes evaluates the "use-as-is" disposition.

Revision 1 made minor editorial changes and clarified points within the body of the evaluation. Changes are marked via revision bars.

EVALUATION:

2.1 Describe the safety function(s) or safety support function(s) of the SSC. As a minimum the following should be addressed, as applicable, in describing the SSC safety or safety support function(s):

- Detailed inspections have been performed to locate the dryer part(s). The lost part(s) were not found. Based upon the detailed inspection results and engineering judgment documented in EC Evaluation 345951, the lost part(s) are not located in the steam path of the reactor vessel, the steam lines, and associated systems but rather entered the Recirculation System. Therefore, the safety functions of the MSIV's, ADS, HPCI, and RCIC are not impacted by this lost part(s). The Core Spray System has its own injection header and is not affected by any potential migration path of the lost part(s). The safety functions that could be impacted by the lost part(s) are limited to the Reactor Vessel lower head area (including the reactor pressure boundary), the fuel core, the Standby Liquid Control System (SLCS) injection sparger and associated instrumentation, and the Recirculation System (including the systems taking a suction from the Recirculation System, RWCU and RHR-Shutdown Cooling). Other safety functions being evaluated are:

  1) Fuel bundle flow blockage and fuel damage due to overheating of the fuel cladding,
  2) Control rod operation,
  3) Corrosion or adverse chemical reactions with other reactor materials,
  4) Interference with the Reactor Water Cleanup (RWCU) System and
  5) Interference with the Nuclear Boiler or Neutron Monitoring Instrumentation.

  In addition, there are operational concerns with:

  1) Potential fuel fretting,
  2) Potential blockage of the reactor vessel bottom head drain, and
  3) Potential for impairment of the Recirculation System performance.

  GE Report GE-NE-0000-0023-5200-R0 evaluated these aspects and determined that the presence of the lost part(s) will not compromise safe reactor operation.

- Does the SSC receive/initiate an RPS or ESF actuation signal?

  The lost part(s) has the potential to affect several SSC's that receive a RPS or ESF actuation signal. These include:
  1) Containment isolation valves for Main Steam (the MSIV's), RHR, and RWCU;
  2) CRD System, preventing rod insertion;
  3) ADS, limiting flow to relief valves;
  4) HPCI and RCIC, preventing or limiting steam flow to the turbines.

  In the steam cases, no evidence of lost part(s) were found in the inspections of the steam path locations where the missing part(s) could have gone, thus the focus of this evaluation will be on the RHR, RWCU, Recirculation, and CRD Systems.

- Is the SSC in the main flow path of an ECCS or support system?

  YES. The Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) System could be impacted if the Recirculation Pump discharge valve could fail to close, a RHR pump were damaged, or LPCI piping is obstructed preventing adequate flow. The RHR System has not been impacted, because of the geometry of the RHR lines and the fact that the system was isolated when the missing part(s) became dislodged. This conclusion is further addressed in section 2.2.

- Maintain reactor coolant pressure boundary integrity?

  The lost part(s) have the potential to fret reactor pressure boundary components at the lower head region and recirculation piping if given the proper conditions. This fretting potential is further evaluated in section 2.2 below and found to not be a significant risk.

- Shutdown the reactor?

  The lost part(s) have the capability of affecting the control rods and their associated drive systems depending upon the size of the lost part(s). Alternately, the Standby Liquid Control System can inject boron to shutdown the reactor if the control rods are not effective. The evaluation in section 2.2 and the GE Lost Parts Analysis show that there is no effect on the CRD System from the lost part(s).

- Maintain the reactor in a safe shutdown condition?

  After the reactor is shutdown, the lost part(s) will have no further effect on reactivity. The lost part(s) have no effect on the Standby Liquid Control System (SLCS) and its capability. The SLCS sparger is constructed of stainless steel and impingement of the lost part(s), which is which is stainless steel plate, would be of minimal to no consequence because of the relative small mass of the part(s), and the relatively lower velocities in this region. Therefore, the instrumentation functions of the SLCS sparger would also not be affected.

- Prevent or mitigate the consequences of an accident that could result in offsite exposures comparable to 10 CFR 50.34(a)(1) or 10 CFR 100.11 guidelines, as applicable.

  The lost part(s) will not change any accident or accident response. The potential to impact ECCS are evaluated in section 2.2 below. The conclusion is that the capacity or reliability of those systems is not impaired.

- Does the SSC provide required support (i.e., cooling, lubrication, etc.) to a TS required SSC?

  The lost part(s) could affect Tech Spec required SSC's, such as RHR pumps. Where that potential exists, it has been evaluated in this Operability Assessment and the systems will still be able to perform their design function.

- Is the SSC used to provide isolation between safety trains, or between safety and non-safety ties?

The lost part(s) will not prevent any isolation functions between safety system divisions or between safety and non-safety systems. The potential impact on containment isolation valves, such as RWCU and the MSIV's is addressed in section 2.2.

- Is the SSC required to be operated manually to mitigate a design basis event?

  The lost part(s) do not cause or prevent any SSC from being operated manually. The lost part(s) do not require new manual actions where automatic actions would have functioned previously.

- Have all safety functions described in TS been included?

  YES, the safety functions that could be impacted by the lost part(s) have been included in this evaluation. These safety functions include LPCI, fuel integrity, control rod operations, chemical reactions with other materials, nuclear boiler or nuclear instrument monitoring systems, RHR pumps and heat exchangers, and reactor internals. Section 2.5.1 of this evaluation lists the Technical Specification sections that were referenced.

- Have all safety functions described in UFSAR or pending revisions been included?

  Yes, the safety functions described in the UFSAR that could be impacted by the lost part(s) are included in this evaluation. Section 2.5.2 of this evaluation lists the sections of the UFSAR that were referenced.

- Have all safety functions of the SSC required during normal operation and potential accident conditions been included?

  Yes, see the response to section 2.2.

- Is the SSC used to assess conditions for Emergency Action Levels (EALs)?

  The lost part(s) do not prevent any condition assessment for EAL determination. The potential for the lost part(s)s to impact nuclear instrumentation or instrument lines used for the measurement of reactor pressure, level, or other safety significant instrumentation is evaluated in section 2.2.

2.2 Describe the following, as applicable: (a) the effect of the degraded or nonconforming condition on the SSC safety function(s); (b) any requirements or commitments established for the SSC and any challenges to these; (c) the circumstances of the degraded/nonconforming condition, including the possible failure mechanism(s); (d) whether the potential failure is time dependent and whether the condition will continue to degrade and/or will the potential consequences increase; and (e) the safest plant configuration, including the effect of transitional action:
In each of the paragraphs below, a separate discussion is presented to address the possible affected system components / functions through describing the characterizations of the items listed in 2.2 above.

a) Low Pressure Coolant Injection (LPCI) of the Residual Heat Removal (RHR) System

   If a single lost part(s) passed through the Recirculation Pump, it could lodge in the recirculation discharge line. If it did lodge there, it might affect the safety function of the

recirculation pump discharge valve. This valve is required to close during a LOCA to allow ECCS injection to be effective (LPCI loop select logic function). The 1-0202-5B valve was Votes tested (acceptable) after the Unit was shutdown for this dryer repair. Both the 1-0202-5A/B valves have been stroke timed (acceptable). There were no indications of hot spots (no variations in dose rates) in the discharge piping that might indicate the presence of the dryer part(s). The recirculation pump has been run and it has normal vibrations. All of these items support the conclusion that the missing part(s) are not currently in the valve or the discharge piping. Since the lost part(s) are smaller than these valves, significant obstruction is not possible. The impact marks on the impeller indicate that the missing piece has broken up into multiple smaller parts and would not hang-up on a pipe fitting or stay in the much larger valve. If a lost part(s) were to prevent the recirculation discharge valve from fully closing, the loss of LPCI flow would delay the core re-flooding, resulting in an increase in the Peak Cladding Temperature (PCT). This would only be an issue for the DBA accident with the Diesel Generator failure scenario. For smaller breaks, or LPCI injection scenario, the discharge valve closure function is not required to maintain acceptable PCT. Based on bounding evaluations, using realistic models and assumptions with no credit for LPCI or discharge valve closure, one Core Spray System is sufficient to keep the PCT below 2200 °F. Therefore, the lost dryer part(s) does not pose a significant safety hazard.

MO1-0202-6A/B and 1-0202-9A/B are given a closed signal during LPCI Loop Select. However, only the 9A is open during normal operation. The remaining valves are operated infrequently, and the 1-0202-6A/B and 9B are normally closed and have remained closed since this failure was detected on the Unit 1 Dryer. These valves are only tested during refuel outages. Therefore, the Recirculation Loops will remain isolated, and there is no effect on the LPCI Loop Select function. Consideration was given for a line break between MO1-0202-6A/B. However, the FME concern is not applicable to this issue since it would be no different than a pipe break anywhere along the A Loop Recirculation Discharge.

b) Fuel bundle flow blockage and fuel damage due to overheating of the fuel cladding

Part(s) smaller than the 3.315 inch jet pump nozzle could pass into the lower plenum via the recirculation loop. Jet pump suction flow could allow a part(s) that are up to 7.0 inches wide to pass into the lower plenum. The results of the analyses to determine the orifice and lower tie-plate blockage necessary to cause boiling transition for Quad Cities at Extended Power Uprate (EPU) conditions with GE 14 and Atrium 9B fuel are contained in the GE Lost Parts Analysis. If the part(s) lateral dimension is small enough to enter the orifice, the flow blockage of the lower tie plate is not enough to cause a dryout condition of the fuel element. The lower plenum geometry and lift force on the part preclude the part(s) from blocking a side entry orifice. Peripheral fuel bundles have bottom entry orifices, however, even if an orifice is completely blocked, adequate cooling from reverse flow from the lower tie plate bypass flow holes and fuel channel clearances will occur. The GE report concluded that there is no safety concern for the reasons noted above.

c) Control Rod Operations

Per the GE Lost Parts Analysis, sect. 2, FME that is greater than 0 .5" cubed are too large to interfere with control rod operation regardless of the migration path they traveled. Even if part(s) of the dryer fell on the top of a withdrawn control blade, it would not cause binding. If any part(s) this size or smaller were to enter the guide tube or rest on the CR blade-to-fuel support piece clearance space, binding would still not occur due to the

extremely high pressure difference acting on the drive from the hydraulic control unit (HCU). Therefore, no potential for interference with the safety functions of the control rods exist.

d) Corrosion or adverse chemical reactions with other reactor materials

In BWR NSSS systems, the low conductivity demineralized water used does not contain significant quantities of ions that would accelerate galvanic corrosion effects. Stainless Steel (such as this lost part(s) construction of 304 stainless steel) has been used in many of the components in the reactor pressure vessel and throughout the NSSS and nuclear plant Recirculation Systems for many years without corrosion or adverse chemical reaction with other reactor materials. As an example, the dissimilar welds on the stainless recirculation inlet and outlet nozzle safe ends are directly welded to low alloy or carbon steel. These welds have seen hundreds of reactor operating years of service and undergone frequent inspections with no reports of accelerated general corrosion. Therefore, galvanic corrosion is not a concern. It is concluded that there will be no significant corrosive or adverse chemical reactions with other reactor materials.

e) Interference with Reactor Water Cleanup (RWCU) System

Inspections were performed on the "B" Recirculation Suction piping with no indications that the part(s) went into the SDC suction pipe. The RWCU suction pipe is a 6" line coming off the top of the 20" SDC suction pipe, and there is no impact evidence that the missing part(s) went through the SDC suction pipe to the RWCU system.

If the part(s) was not broken (6.5 X 9.0 inches), it is not conceivable that it could enter the 6" RWCU line. Therefore, only migration of the smaller lost part(s) could potentially interfere with the redundant RWCU isolation valves. The RWCU System is normally in operation and these valves are normally open but close during an accident such as a LOCA or HELB outside containment. The smaller lost part(s) could be carried by the RWCU flow to the isolation valves, but this is not expected due to the piping configuration and operating characteristics of the system. The RWCU System connects with a vertical pipe to the top of a horizontal section of the RHR shutdown cooling piping.

During normal operation, the flow velocity through the RHR line due to operation of RWCU is expected to be too low to carry the smaller part(s) through the RHR piping and up into the RWCU piping due to the large diameter of the RHR piping (20") versus the small diameter (6") of the RWCU piping. Changes in flow characteristics after reactor shutdown (shutdown cooling in operation) would have no further effect on part(s) migration since any small parts would be swept into the RHR System and not RWCU based on comparison of flows (~4000 gpm RHR SDC vs. ~400 gpm RWCU).

In the event that a smaller part(s) gets trapped in a recess area inside one of the RWCU isolation valve bodies, it could prevent the complete operation of the valve. If the part(s) do not hang-up on pipe fittings, then they are unlikely to stay in the gate valve used for isolation. Therefore, it is not expected that the smaller part(s) capable of entering the system will interfere with the safety or normal operation of the RWCU valves such that closure would be prevented or impaired. These valves have been used as isolation for repairs as well as stroked and timed during the outage and there is no evidence of foreign material in these valves. The RWCU system is on and running with no indication of any foreign material affecting system performance.

If the smaller part(s) were discharged into a RWCU System pipe leading to the associated heat exchangers, they could cause minimal blockage of the heat exchanger tubes. Given the surface area of the missing piece, the heat exchanger would continue to function as designed. If efficiency were impacted, existing instrumentation would indicate the problem. If the part(s) did pass through the heat exchanger, they might also pass to the RWCU Filter-Demineralizer where they would be trapped. The small metal part(s) would lie in the bottom of the Demineralizer, because the top tube sheet Demineralizer has internal velocities that could not lift the metal to the filters. The material would then be washed out of the Demineralizer to Radwaste resin settling tanks. The small metal part(s) would pose no special safety hazards in the resin storage tanks.

The RWCU system also takes suction from the reactor vessel bottom head. Some of the lost part(s) may enter the reactor lower plenum. An accumulation of smaller part(s) could partially block the bottom head drain line opening, resulting in a reactor water conductivity increase. In addition, temperature differences between the reactor vessel bottom head and the recirculation loops would need to be monitored if an idle Recirculation system loop would be started. This is not a concern because procedural guidance for monitoring the temperature of Recirculation Systems loops and Reactor vessel head temperatures already exists and reactor water conductivity is monitored by Chemistry via sample on a daily basis and by Operations via the conductivity recorder on the 901-4 panel.

f) Interference with the Nuclear Boiler or Neutron Monitoring Instrumentation

If there is only a single lost piece, it will not migrate to the Nuclear Boiler Instrumentation, due to an absence of flow through the pressure and differential pressure instrument lines associated with vessel water level and recirculation flow. There would be no flow to carry the part(s) into the lines and therefore it is not reasonable to assume that the part(s) would fall into the horizontal small bore instrument sensing line penetrations.

The smaller lost part(s) could migrate into the lower plenum and with normal reactor flow; the part(s) could impinge upon the Neutron Monitoring System (SRM/IRM/LPRM) guide tubes. The SLCS sparger also provides the instrument tap for several instruments:

- Core plate differential pressure
- CRD high pressure side control and cooling water flow differential pressure.
- Jet Pump high pressure side for differential pressure.
- Total Jet Pump differential pressure high side.

The smaller lost part(s) could migrate to the lower plenum and impinge upon the SLCS sparger. However, due to the relative small size of these part(s), there would be no significant consequence of the impingement. With the exception of the Nuclear Instrument guide tubes, components at the bottom of the vessel are thick and not expected to wear through from fretting. Therefore, there is no safety concern for Nuclear Boiler or Neutron Monitoring Instrumentation due to the lost part(s). The Nuclear Instrument guide tubes are vulnerable to fretting but this is not considered likely because of the lower flow velocities in this area. Even if a guide tube were to fret through wall, the tube does not make up part of the pressure boundary in the lower plenum.

g) Potential Fuel Fretting

One or more of the smaller lost part(s) may pass through a lower tie-plate of the Atrium 9B fuel and could wear a hole in the fuel cladding. The smallest assumed part(s) (see

GE Lost Part(s) Analysis, sect. 2) would not enter the lower tie-plate of any GE 14 fuel or the upper tie-plate of either fuel type.

If fuel cladding leakage did occur, it would be detected by the Offgas System so that appropriate actions could be taken to maintain the off-gas radiation release within acceptable limits. All applicable TS actions would be carried out as required. Additional information and recommendations are contained in SIL 552, "Fuel Failures Caused By Metal Debris."

There is some possible operational concern for the smaller lost part(s) to cause fuel fretting, but this does not constitute a safety issue because this situation is analyzed. In addition, any fuel fretting would be self-revealing from off-gas radiation monitoring.

h) RHR Valves, Pumps, and Heat Exchangers

The RHR suction line (a 20" line) comes off the side of a vertical run of the larger Recirculation System pipe (28" line) upstream of the Recirculation Pump suction. Empirical evidence from inspection results has shown that the part(s) did not enter the RHR suction line because impact marks were found on the Recirculation Pump impeller with no evidence of impacts in the SDC suction header. The RHR isolation valves (1-1001-47&50) are also redundant and closed during normal operation. Also, the isolation valves for RHR are required to be closed during a drain down evolution and normal power operation. The redundancy of isolation valves makes the failure of both valves very unlikely. The existence of these loose part(s) does not make the failure of these valves more likely, because the foreign material didn't have an opportunity to enter the system. Therefore, it is not expected that the lost part(s) will interfere with the safety or normal operation of the RHR isolation valves such that closure would be prevented or impaired.

Smaller lost part(s) could migrate and pass through the RHR pumps without causing any damage. Normal flow rate testing the RHR pumps will provide ample assurance that the part(s) are not lodged in the RHR System and that a problem with the pumps or valves will go undetected. During the outage, the only pump that was operated in SDC mode of operation was the 1A RHR pump. This pump performed normally during the SDC operation. No other RHR pumps were exposed to the lost part(s) and therefore are not affected by any lost part(s) evaluation.

Upon passing through the pump, the part(s) would be discharged into the system pipe leading to the heat exchangers, where it would not pass through but would stop and cause minimal blockage. Only the "A" heat exchanger would be affected since only the "A" loop of RHR using the 1A RHR pump was used in the SDC mode of operation. The "B" heat exchanger would be unaffected by any lost part(s).

Therefore, there is no operational concern for the operation of the RHR pumps or heat exchangers.

i) Potential for Impact Damage on Reactor Internals

All the part(s), whether large or small, could land in the reactor annulus, where they could then migrate into the Recirculation System and from there into the lower plenum. Since there are indications that the lost part(s) passed through the "B" recirculation pump, it will then have to pass through the jet pump nozzle to enter the lower plenum. This will limit

the size of the part(s) in the lower plenum to a size equal to or less than the 3.315" jet pump nozzle.

If it is a relatively large part(s), the high downward velocity will tend to keep it on the bottom. The primary factor affecting this will be the radial component of the velocity. The radial velocities are approximately 7 ft/sec at the periphery and drop significantly as it nears the center and bottom of the vessel. This will cause the part(s) to be initially pushed toward the vessel center where they would be expected to remain. The part(s) could then lodge in the shroud support legs or between 4 CRD stubs and an in-core instrument housing, where there is essentially no clearance.

With the exception of the Nuclear Instrument guide tubes, components at the bottom of the vessel are thick and not expected to wear through from fretting. The Nuclear Instrument guide tubes are vulnerable to fretting, but this is not considered likely because of the low fluid velocities in this area. Even if FME did fret through-wall on a nuclear instrument guide tube, this damage does not compromise the pressure boundary in the lower plenum and would be immediately detected by the instrumentation response.

The impingement by these part(s) on the shroud or its' hardware, recirculation lines, jet pump assemblies, core spray piping, and other reactor internals would be of minimal consequence due to their small size of the missing part(s) of metal. If any performance deterioration were to occur, it could be monitored so that corrective actions could be taken, as necessary.

Therefore, there is no expected damage from the part(s) and no impact on the reactor pressure boundary or the ability to maintain 2/3 core coverage following a LOCA.

j) Potential for FME To Travel Up the Recirculation Sample Line

The 1-0251- ¾"-A HRSS Recirculation Sample line may also be affected by this issue. AO 1-0220-44/45 provides containment isolation for that line and are normally open valves. This ¾" line taps off the center of the 22" Recirculation Discharge pipe on the horizontal run of the pipe. The FME would be required to travel 7' in the vertical direction to reach the 1-0220-44 valve and another 7' in the vertical direction to reach the penetration/ 1-0220-45 valve. This highly unlikely given the fact that the line sees flow only when a sample is being taken. The flows in this line are significantly smaller when compared to the high velocity flows in the 1B Recirculation Discharge line. Therefore, this issue does not affect the Primary Containment Isolation function of the HRSS Sample line.

In the cases detailed above, the lost part(s) were determined to have minimal to no impact on the potentially affected systems, structures, and components.

		YES	NO
2.3	Is SSC operability supported? Explain basis (e.g., analysis, test, operating experience, engineering judgment, etc.):	[X]	[ ]

The operability of the steam dryer, reactor vessel components, and adjoining piping systems is supported. This evaluation demonstrates operability through analysis of the potential effects of the lost part(s) on plant operation during normal and accident conditions and uses the analyses performed during CR 168367, "Unit 1 Steam Dryer Potential Non-conformance"; EC 343900, "Quad Cities Unit 1 & Dresden Unit 2 & 3 Steam Dryer Acceptability for Operation at EPU Power Levels"; CR 111976, "Moisture Carryover Preliminary Analysis – Unit 2"; CR 158145 "Unit 2 Moisture Carryover Increase"; and GE Lost Parts Analysis (Reference 20) in conjunction with EC 345951, "Document the Result of Steam Dryer Loose Parts Inspection and Retrieval Effort" evaluations as the bases.

The lost part(s) of the dryer is a 6.5 by 9.0 inch by ½-inch thick piece of grade 304 stainless steel that may exist in any of the following states:

- A single large piece
- A few smaller part(s)
- Multiple small pieces(s).

Therefore, the effects of both large and small part(s) will be addressed.

<u>Lost Part(s)s Analysis (Reference EC 345951)</u>
Detailed inspections were performed on the following to attempt to locate the missing part(s) in the following locations:

- the Reactor Steam Dryer
- the Reactor Moisture Separator
- the annulus area
- top of core
- jet pumps
- instrumentation sensing lines
- shroud tie box
- feedwater sparger
- Core Spray sparger
- A&B Recirculation System piping from the vessel to Recirculation Pump suction
- 1B Recirculation Pump impeller
- SDC line suction from its interface with the Recirculation System to the inboard isolation valve
- the inlet to RWCU line
- C&D Main Steam Lines from the vessel to the Main Steam Stop Valves
- the number 1,2,3 and 4 Stop Valves
- the Main Steam Bypass Valves
- the A&B Main Steam Lines from the vessel to the flow venturi.

This effort included GE Firefly inspections, boroscope inspections, pole-mounted cameras, robotic crawlers, radiography, dose rate analysis and a review of plant performance parameters.

Neither the piece nor any portion of it was located in these inspections. No damage or indications of movement of the part(s) were identified during this inspection except for some impact marks that were identified on the 1B Recirculation Pump impeller. The effect of the impact marks on the

impeller were evaluated as part(s) of CR# 188260. A review of Recirculation and Jet Pump flows did not provide any indication that missing part(s) had blocked or reduced flow downstream of the Recirculation Pump.

The Station inspections to locate the lost part(s), the Lost Parts Analysis, and the physical evidence of the Recirculation Pump Impellor impact marks lead the Station to conclude that the most likely location of the part(s) is in the lower plenum.

The reference General Electric Company report (GE-NE-0000-0023-5200, R0) evaluates the 6.5 by 9.0 inch by 0.5 inch thick piece of grade 304 stainless steel lost part(s) and analyzed the following issues both as a large piece and as smaller pieces:

- Main Steam Isolation Valve function,
- Fuel Bundle blockage,
- Control Rod operation,
- Corrosion or adverse chemical reaction with other reactor materials,
- RWCU or RHR isolation valves,
- Nuclear Instrumentation,
- Damage to the fuel due to fretting,
- RWCU or RHR Pumps, RWCU or RHR Heat Exchangers and RWCU Filter Demineralizers,
- Blockage of the Reactor Vessel Bottom Head Drain,
- Impairment of Recirculation System performance,
- Reactor internals,
- HPCI and RCIC operation and
- Impacts on transient and accident response.

The report concluded that safe reactor operation would not be compromised with the lost part(s) in the reactor vessel. There was no safety concern with fretting of components in the lower plenum, flow blockage to the fuel bundles, interference with control rod operation, corrosion or adverse chemical reaction with other reactor materials, interference with Nuclear Instrumentation, or interference with Main Steam, RWCU, or RHR isolation valves. There is no operational concern relating to interference with operation of the HPCI or RCIC Systems, RWCU or RHR pumps, RWCU or RHR heat exchangers, or the RWCU Filter Demineralizers.

The report states that there is some operational concern for the potential lost part(s) to cause fuel fretting, however, any fuel cladding leak would be detected by the Offgas System so that appropriate actions could be taken to maintain any radiation release within acceptable limits.

There is a second possible operational concern with flow blockage of the reactor vessel bottom head drain. If some of the lost part(s) partially blocked the bottom head drain line opening, reactor water conductivity may increase. In addition, temperature differences between the top and bottom of the reactor vessel and Recirculation System loops would need to be monitored for delta-T concerns if startup of an idle loop is desired. This is not an issue in that QCOP 0202-02, "Reactor Recirculation System Startup", provides sufficient guidance with regard to delta-T requirements when staring idle loops. Additionally, Chemistry monitors reactor water conductivity per the guidance provided in CY-AB-120-100, "Reactor Water Chemistry" thus, no additional guidance is required for this potential concern.

The last concern is regarding the movement of lost part(s) into the Recirculation System itself. In this case, the daily Jet Pump operability surveillances will detect any potential degradation due to damage or blockage due to lost part(s). Review of the material condition of the Jet Pump Sensing lines did note that the Jet Pump #7(8) does not provide valid indication of flow. However, the #7(8) Jet Pump sensing line issue is not related to the FME analysis since the predominant indication of

FME travel is through the 1B Recirculation Line, and Jet Pump #7(8) is located on the A Recirculation Loop.

There is no adverse impact to the plant response to accidents and transients from the lost part(s).

Thus, the lost part(s), whether they exist as a single piece or multiple small pieces, have no effect on the safety functions of potentially impacted systems. Other effects that could occur can be discerned by existing monitoring and mitigated.

The GE Report was reviewed and acceptance was documented under EC 345951 with the associated Nuclear Fuels Design Analysis.

Conclusion

With regard to a safety functions specified in section 2.1, there is no safety concern for potential significant fuel bundle flow blockage and consequent fuel damage due to clad heat-up; there is no potential for interference with the safety functions of the control rods; there is no potential for corrosion or adverse chemical reaction with other reactor materials; there is no potential for interference with the RWCU System pumps or heat exchangers, and there is no potential for interference with the Nuclear Instrumentation. The operational concerns of fuel fretting, potential blockage of the Reactor Vessel Bottom Drain, and potential impairment of the Recirculation System performance are already addressed via existing procedures and surveillances and no new actions are necessary.

Therefore, in either case of small or large lost part(s), there is no effect on reactor operation or accidents or transients and any of the potential operational concerns have sufficient controls in place to allow effective monitoring of any potentially degraded conditions resulting for the lost part(s). Thus, the RPV and associated components are not adversely affected by the lost part(s) and can be considered operable.

Compensatory and Corrective Actions

The actions listed below will be completed under current Operating procedures and instructions and are not considered "additional" compensatory actions. These actions are performed as part(s) of normal plant start-up and operation. One Compensatory Actions is required to monitor the health of the recirculation pump to determine the trend of vibration results. This does not affect the method by which the plant is operated and no 50.59 evaluation for the compensatory action is required. One Corrective Action to perform an inspection of the lower reactor head in Q1R18 and locate and remove the foreign material or evaluate the material inplace. Other actions are occurring as part of routine system operations.

1)Testing the RHR pumps at high flow prior to startup

A review of plant records indicates that the only RHR pump run in the shutdown-cooling mode during Q1F51 was the 1A pump. Given the fact that both the 1B Recirculation Loop and 1A Shutdown Cooling Loops were run for several days, there was sufficient time for any postulated FME to travel to the 1A RHR Pump or the bottom of the RPV. Control room indication shows that this pump was run at approximately 4000 gpm during this time period. The 1D RHR Pump was run during the outage on several occasions to transfer water from the torus to the condenser hotwell. However, the shutdown cooling suction valve for that pump, MO 1-1001-43D, was never opened. As a result, there is no potential for the missing part(s)s to be in the 1D RHR pump. The

1B and 1C RHR Pumps were not run during Q1F51, nor were their Shutdown Cooling Suction Valves, MO 1-1001-43B and MO 1-1001-43C, opened. There is no possibility that part(s) may have migrated to any of the other RHR pumps through the suction header.

2)Daily monitoring of reactor water conductivity

Reactor water samples are currently analyzed daily in accordance with CY-AB-120-100, Reactor Water Chemistry. In addition, continuous monitoring is available in the Main Control Room via the 1-1290-25 Conductivity Recorder on the 901-4 panel.

3)Monitoring of top and bottom reactor vessel temperature prior to idle Recirculation Loop restart

QCOP 0202-02, Reactor Recirculation System Startup currently provides the necessary direction for monitoring for differential temperature between the top and bottom of the reactor prior to starting an idle recirculation loop. Failure to meet these parameters would prevent idle loop startup at power and would require plant shutdown. This is an operational concern and not a safety concern.

4)Monitoring of Recirculation System flow parameters

This monitoring is currently performed daily via performance of QCOS 0202-07, Jet Pump Flow Distribution Comparison.

5) An inspection of the lower plenum will be performed in Q1R18 to locate and remove the FME or an accept-as-is disposition will be performed through the 50.59 processes to evaluate the part(s) in the vessel prior to operation after Q1R18.

6) Vibration measurements will be taken of the 1B Recirculation Pump as a follow-up action on the evaluation of the impact marks on the impeller. This will occur weekly for the first two months then the data will be analyzed and the collection period re-evaluated.

If 2.3 = NO, notify Operations Shift Management **immediately**.

If 2.3 = YES, clearly document the basis for the determination.

		YES	NO
2.4	Are compensatory and/or corrective actions required?	[X]	[ ]

If 2.4 = YES, complete section 3.0 (if NO, N/A section 3.0).

2.5 Reference Documents:

2.5.1 Technical Specifications Section(s):

TRM 3.4.a "Structural Integrity"

Technical Specifications Sections:

3.1 Reactivity Control Systems
3.2 Power Distribution Limits
3.3 Instrumentation
3.4 Reactor Coolant System, including:
3.4.1 Recirculation Loops Operating
3.4.2 Jet Pumps
3.4.3 Safety and Relief Valves
3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown
3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown
3.5 Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System
3.6.1.3 Primary Containment Isolation Valves

2.5.2 UFSAR Section(s):

3.9.5	Reactor Pressure Vessel Internals
4.5	Reactor Materials
5.0	Reactor Coolant System and Connected Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.1	Reactor recirculation System
5.4.1.1.3	Jet Pumps
5.4.6	Reactor Core Isolation Cooling System
5.4.7	Residual Heat Removal System - Shutdown Cooling and Other Functions
5.4.8	Reactor Water Cleanup System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
7.6	Core and Vessel Instrumentation
9.3.5	Standby Liquid Control System
15.0	Accident and Transient Analysis

2.5.3 Other:

References:

1. NRC Generic Letter 91-18, Revision 1
2. NRC Inspection Manual, Part(s) 9900, Technical Guidance
3. GE-NE-0000-0005-5755-01-01 Revision 1, "Steam Flow Imbalance and Moisture Content Assessment for Quad Cities Generating Station Unit 2, October 2002
4. GE-NE-T2300700-17-17-01, "Lost Part(s)s Analysis for Potential Steam Dryer Lost Part(s)s in Quad Cities Generating Station Unit 2" June 18, 2002
5. GE-NE-B13-02098-00-08P, Revision 1, "Dresden and Quad Cities Steam Dryer Modification for Extended Power Uprate Safety Evaluation"
6. NSA 02-437, "QC Lost Part(s) – Instrument Tubing," DRF T23-00700-17-17-05, July 19, 2002
7. GENE-0000-0018-1060-R0, "Failure Modes and Effects Evaluation for Steam Dryer Components" for Dresden Units 2 and 3 and Quad Cities Units 1 and 2, June 2003
8. Letter RS-03-127, Commitments for Resolution of Steam Dryer Degradation Issue, Exelon to NRC, June 27, 2003
9. Evaluation of Potential Main Steamline Flow Blockage for Quad Cities Unit 2, DRF0000-0018-9257
10. SRV Blowdown Analysis for Operation With 2.5% Moisture
11. GE RPT-VISD-Q2-MJBD1, Visual Inspections (VI) of the Steam Dryer at Quad Cities Unit 2, June 2003 (Q2F59)
12. GE RPT-Q2M20-02-JLM6J, Quad Cities, In Vessel Inspection (IVI), during the Unit 2 Summer Maintenance 2002 Q2M20 Outage, July 2002
13. GENE-000-0005-9763-05, Justification for continued operation "As-is" with the observed indications in the 180° V5 weld and the 0° V1 and V3 welds in the steam dryer at Quad Cities Unit 2", July 16, 2002.
14. NSA 02-420, "QC Postulated Lost Part(s) in Steam Separator," DRF T23-00700-17-17-04, July 12, 2002
15. NSA 02-040, "Lost Part(s)s Analysis for Quad Cities Generating Station Unit 1," DRF T23-00700-17-09, January 15, 2002
16. GE-NE-000-0017-6551 – Draft A, "Potential Dryer Strut Lost Part(s)s," June 14, 2003
17. GENE-000-0023-6710-P, "Quad Cities Unit 1 Steam Dryer Failure – Determination of Root Cause", Revision A, DRF 0000-0023-6710, November 2003.
18. FDDR EE1-0123, "Steam Dryer", November 23, 2003
19. GE-NE-0000-0023-0466-JC08, "Quad Cities 1 Steam Dryer Inner Bank Vertical Brace Indications," November 25, 2003
20. GE-NE-0000-0023-5200-R0, "Lost Part(s)s Analysis for Quad Cities Generating Station Unit 1 Steam Dryer Outer Hood (270° Side)," November 26, 2003

Attachments:
1. Attachment A, Quad Cities Unit 1 (Q1F51) Inspection Results as related to Quad Cities Unit 2
2. Attachment B, Picture of impact marks on the impeller

3.0 ACTION ITEM LIST:

If, through evaluating SSC operability, it is determined that the degraded or nonconforming SSC does not prevent accomplishment of the specified safety function(s) in the TS or UFSAR and the intention is to continue operating the plant in that condition, then record below, as appropriate, any required compensatory actions to support operability and/or corrective actions required to restore full qualification. For corrective actions, document when the actions should be completed (e.g., immediate, within next 13 week period, next outage, etc.) and the basis for timeliness of the action. Corrective action timeframes longer than the next refueling outage are to be explicitly justified as part(s) of the OpEval or deficiency tracking documentation being used to perform the corrective action.

Compensatory Action #1: Vibration measurements will be taken of the 1B Recirculation Pump as a follow-up action on the evaluation of the impact marks on the impeller. Monitoring will occur weekly for a period of two months, after which time vibration trends will be evaluated and the collection period reassessed.

Responsible Dept. / Supv.: A8426CMO / Craddick

Action Due: 1st collection due on 12/12/03, then weekly thereafter. Two month evaluation period due on 2/13/04.

Basis for timeliness of action: This will allow collection of data to evaluate trends and determine if there were any detrimental effects on the Recirculation Pump.

Action Tracking#: To Be assigned by CAP.

Corrective Action #1: An inspection of the lower plenum will be performed in Q1R18 to locate and remove the FME or an engineering change will be performed with a 50.59 to leave the part(s) in the vessel prior to operation after Q1R18.

Responsible Dept. / Supv.: A8451NESPR / Wojcik

Action Due: March 31, 2005 / Q1R18

Basis for timeliness of action: This Operability Evaluation Supports operation until the next refuel outage of sufficient duration, Q1R18. This will give proper time for a formal analysis and/or outage planning.

Action Tracking#: To Be assigned by CAP.

4.0 SIGNATURES:
4.1 Preparer(s) [signature] Date 11/28/03
Date
4.2 Reviewer Errol E. Mendenhall Date 11/28/03
(10 CFR 50.59 screener qualified or active SRO license holder)
4.3 Sr. Manager Design Engg/Designee ~~Concurrence~~ [signature] Date 11/28/03
4.4 Operations Shift Management Approval [signature] Date 11-28-03
4.5 Ensure the completed form is forwarded to the OEPM for processing and Action Tracking entry as appropriate.

5.0 OPERABILITY EVALUATION CLOSURE:
5.1 Corrective actions are complete, as necessary, and the OpEval is ready for closure
Date
(OEPM)
5.2 Operations Shift Management Approval Date
5.3 Ensure the completed form is forwarded to the OEPM for processing, Action Tracking entry, and cancellation of any open compensatory actions, as appropriate.

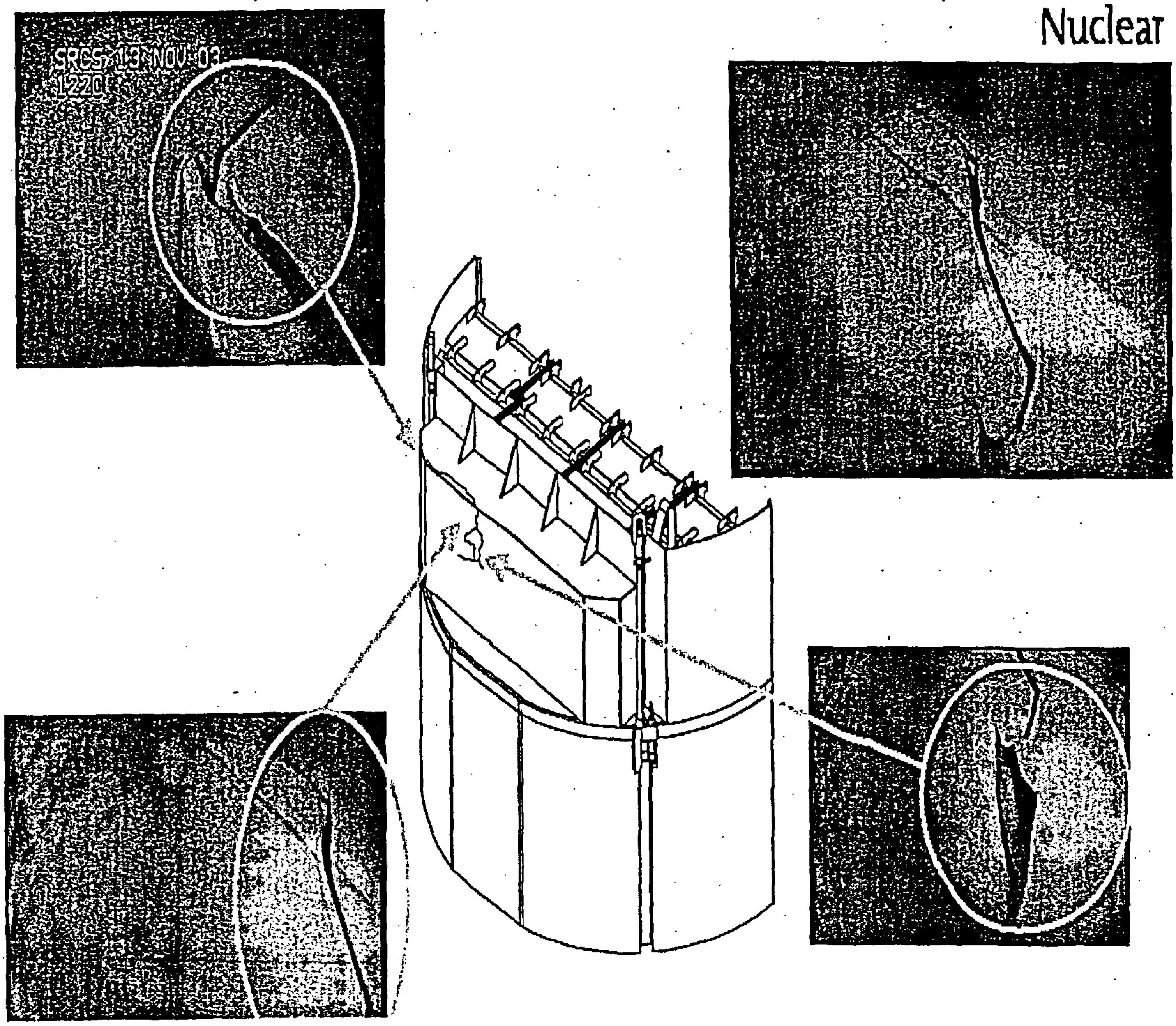
Attachment A

Failure Locations of Quad Cities Unit 1 Steam Dryer

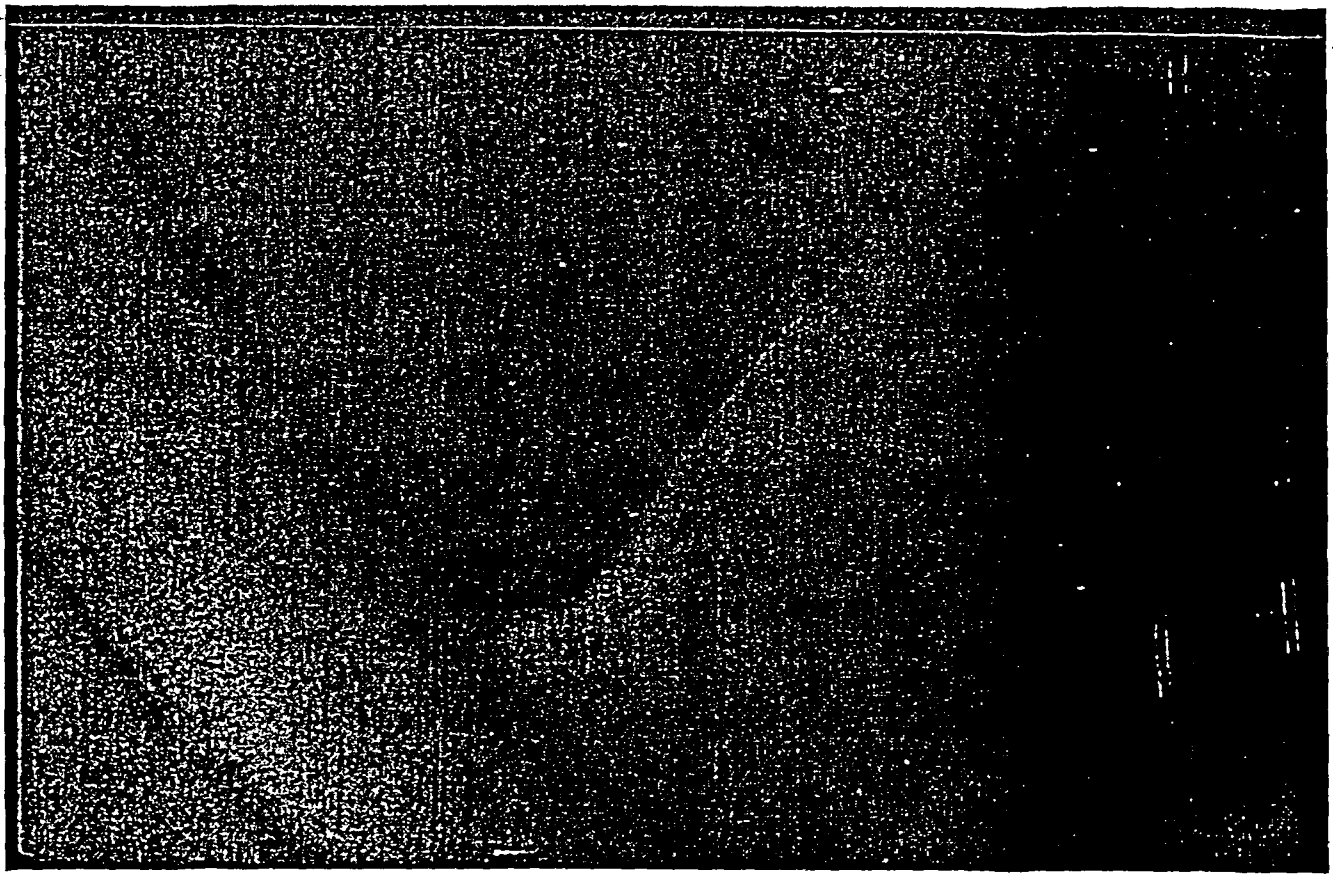
QC1 Dryer Inspections

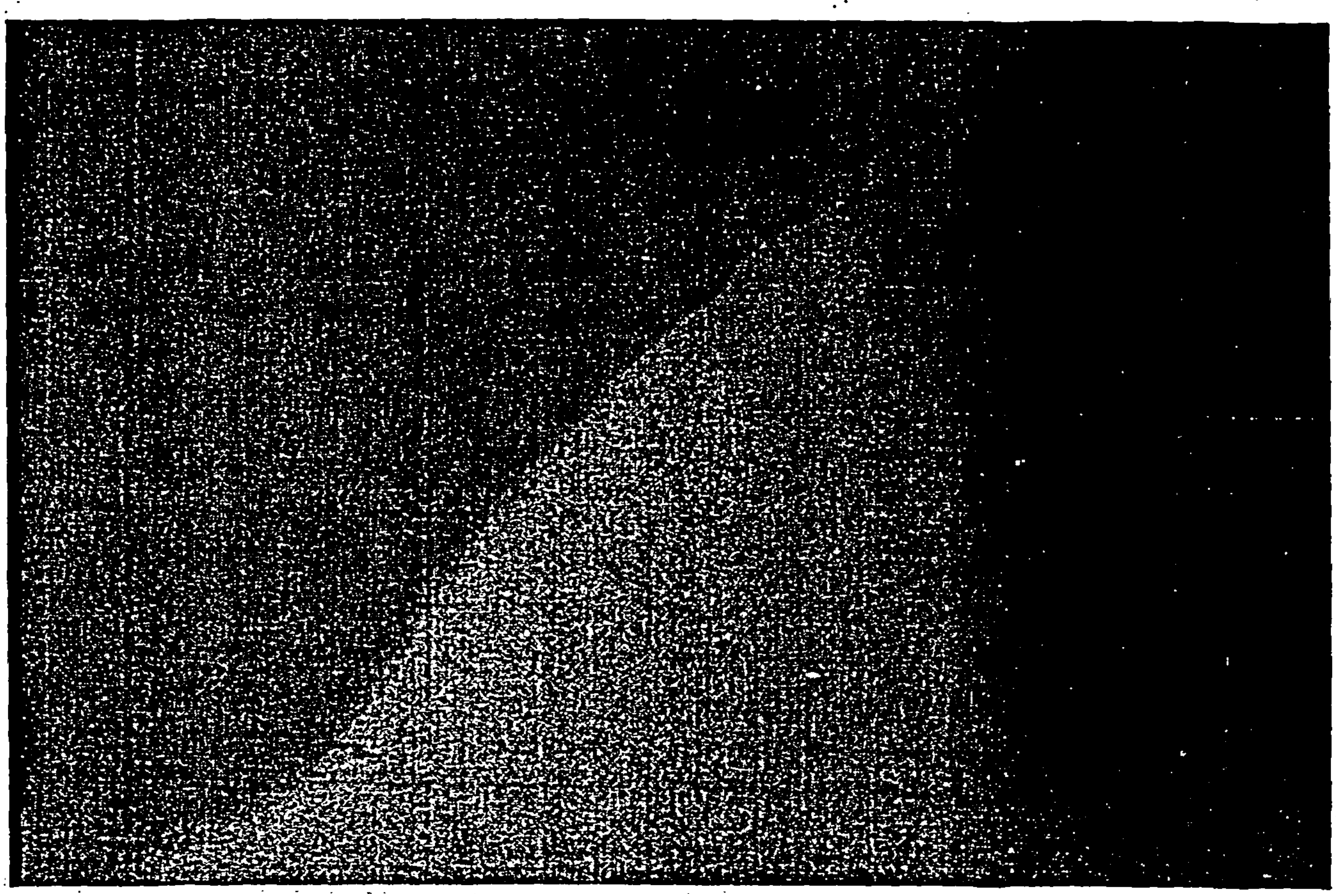
270° Inspection Status

Exelon. Nuclear

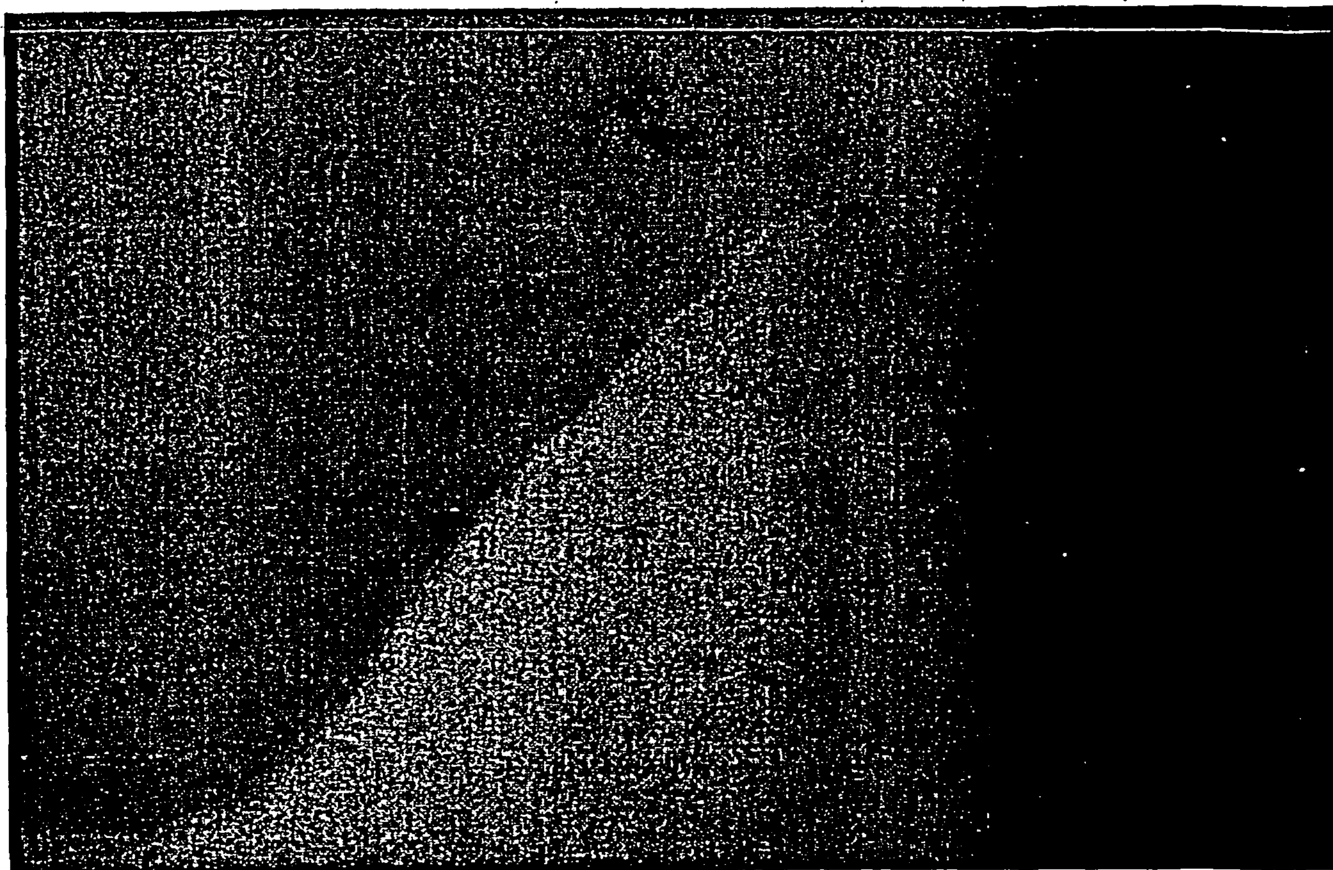


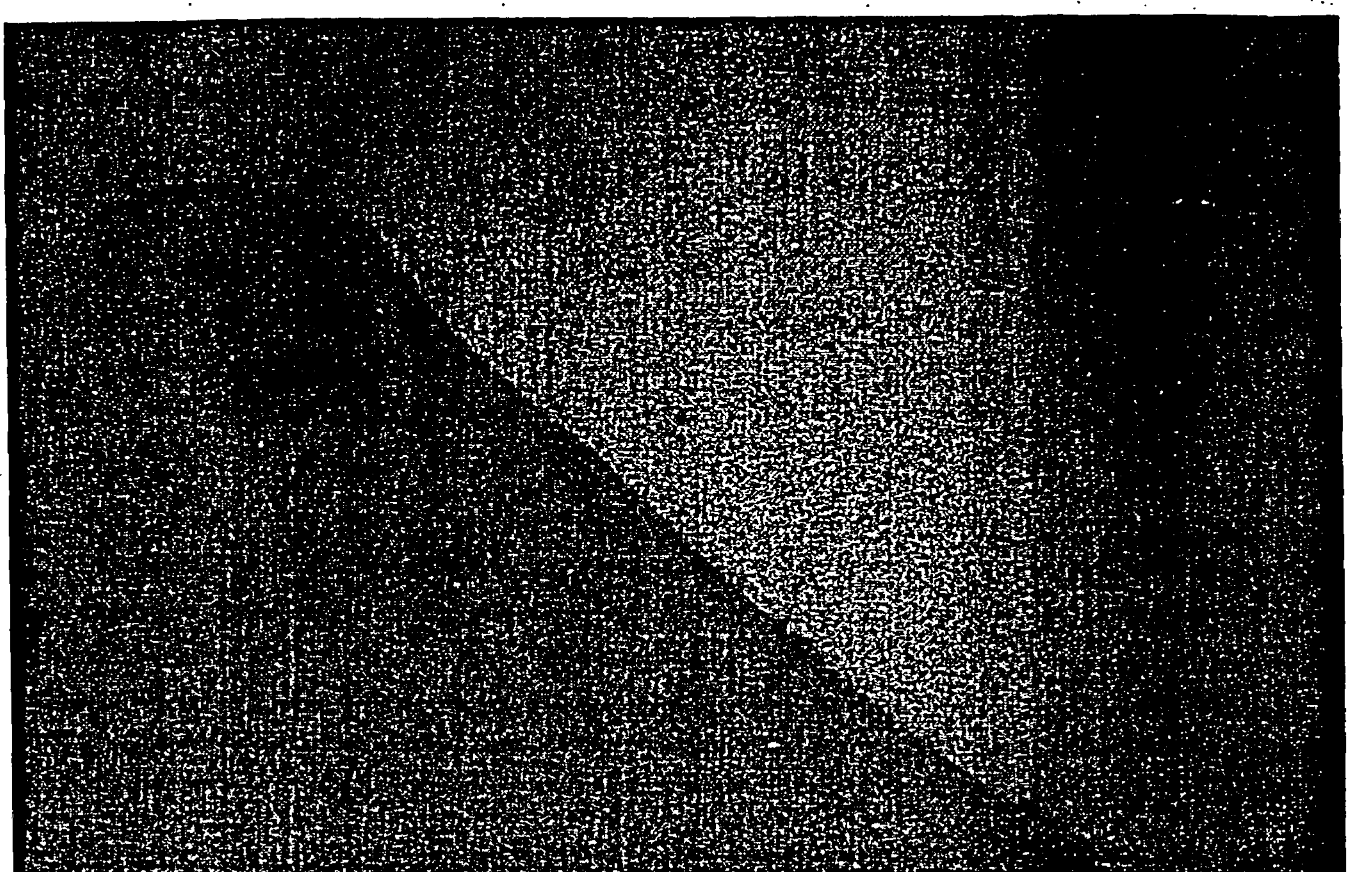
Q1F51 B PUMP FIREFLY PHOTOS





Q1F51 B PUMP FIREFLY PHOTOS





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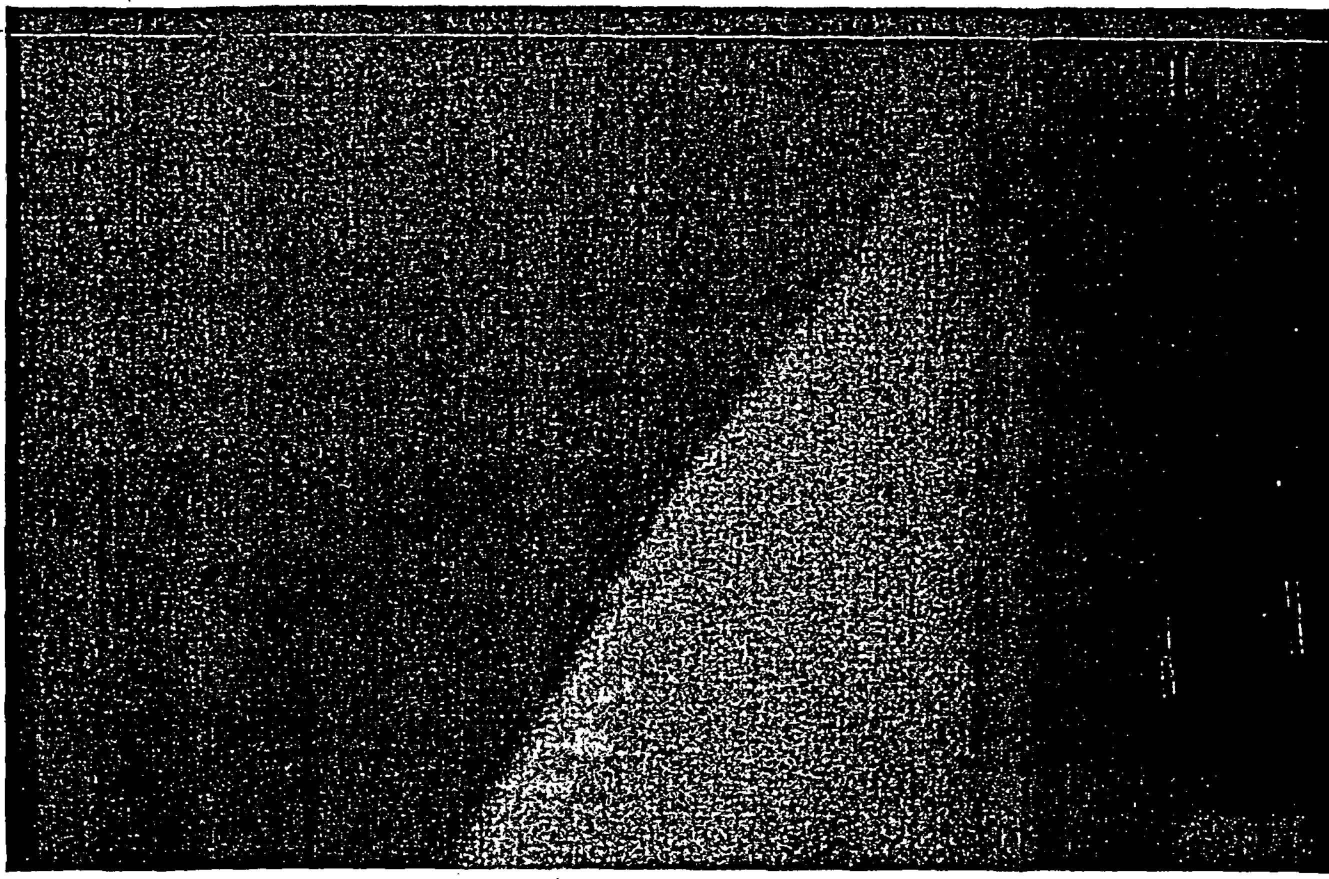

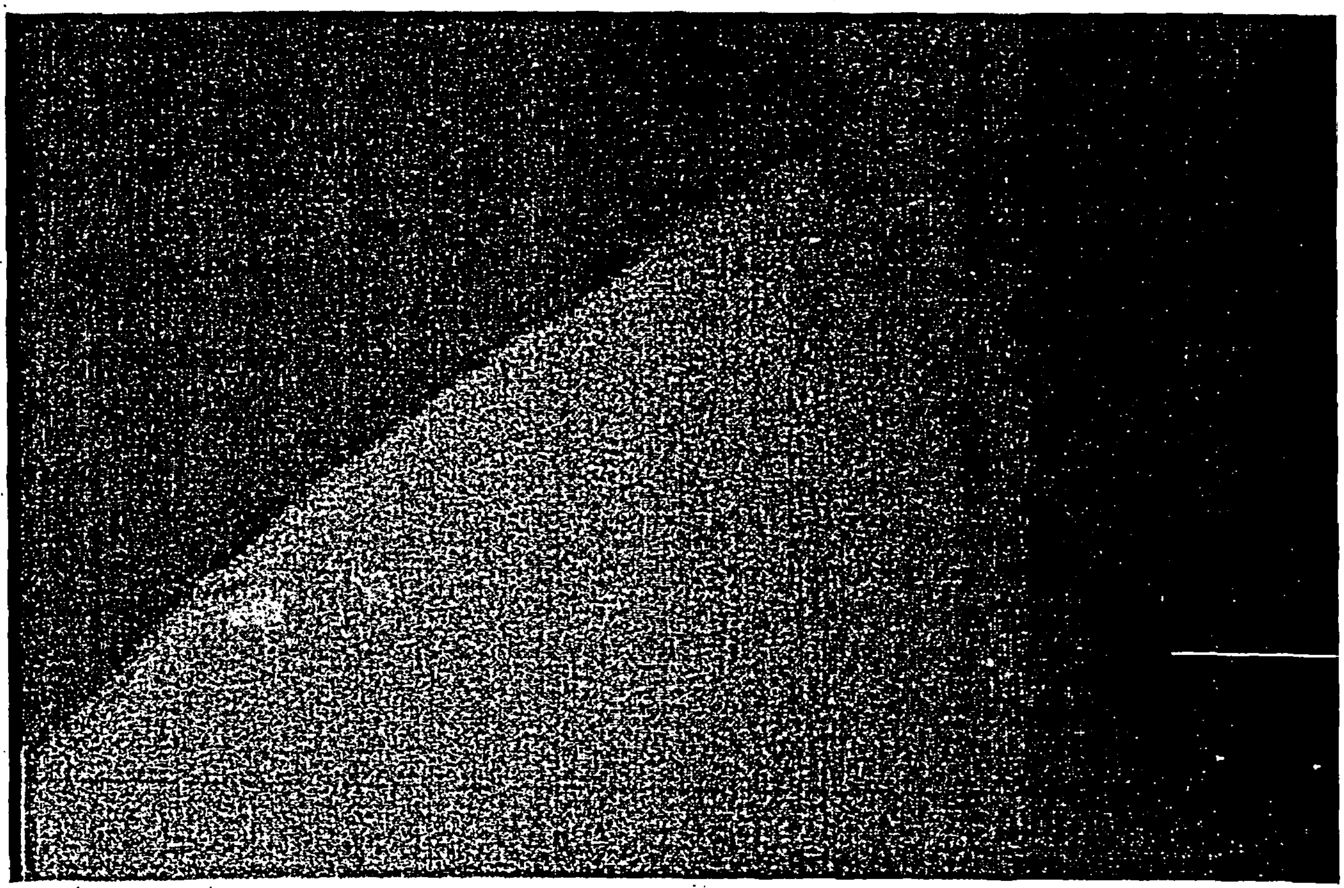

Q1F51 B PUMP FIREFLY PHOTOS