28 SEP 1987

Docket No. 50-219

GPU Nuclear Corporation
ATTN: Mr. P. R. Clark
President
100 Interpace Parkway
Parsippany, New Jersey 07054

Gentlemen:

Subject: Augmented Inspection Team Report No. 50-219/87-29

This refers to the Augmented Inspection Team (AIT) special review conducted by this office on September 11-17, 1987 at the Oyster Creek Nuclear Generating Station, Forked River, New Jersey. This inspection was conducted by a team led by Dr. L. H. Bettenhausen and included a representative from Region I and the Office of Nuclear Reactor Regulation. The team performed a technical analysis of the circumstances surrounding events which resulted in your reported Technical Specification Safety Limit Violation on September 11, 1987 as directed by the AIT Charter (Appendix B). The preliminary findings of this inspection were discussed with you and members of your staff by Dr. Bettenhausen at the conclusion of the inspection.

The areas examined during this special inspection are discussed in the NRC Region I Inspection Report which is enclosed. The areas examined were prescribed by the AIT Charter and were selected to provide NRC management with a comprehensive analysis of the root causes and safety significance of the event.

Issues resulting from this inspection including possible enforcement actions will be subjects for future correspondence.

Sincerely,

original signed by

William F. Kane, Director
Division of Reactor Projects

Enclosure: Augmented Inspection Team Report 50-219/87-29 w/Appendices

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RI:DRP Cowgill/mjd	RI:ES Holody	~~RI:DRP~~ ~~Collins~~	RI:RC Gutierrez
9/23/87	9/25/87	~~9/ /87~~	9/24/87
RI:DRP Kane	RI:DRA Allan	RI:RA Russell	
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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/87-29

Docket No. 50-219

License No. DPR-11 Priority - Category -

Licensee: GPU Nuclear Corporation

P. O. Box 338

Forked River, New Jersey 08731

Facility Name: Oyster Creek Nuclear Station

Inspection At: Forked River, New Jersey

Inspection Conducted: September 11 - 17, 1987

Participating Inspectors:

C. Cowgill, Chief Reactor Projects, Section 1D
W. Bateman, Senior Resident Inspector, Oyster Creek
D. Allsopp, Resident Inspector, Hope Creek
A. Dromerick, Licensing Project Manager, NRR
J. Wechselberger, Resident Inspector, Oyster Creek

Approved By: Lee H. Bettenhausen 9/25/87
Lee H. Bettenhausen, Chief Date
Reactor Projects Branch 1, DRP

Inspection Summary:
Inspection on September 11 - 17, 1987 (Report No. 50-219/87-29)

Areas Inspected:

Technical issues associated with the reported violation of Safety Limit 2.1.E on September 11, 1987 wherein a condition requiring at least two sets of suction and associated discharge valves in the five reactor recirculation loops to be in the full open position was not met.

Results:

See inspection summary.

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1. Summary

On September 11, 1987, maintenance activities in progress at the plant led to the violation of a Technical Specification Safety Limit. The purpose of this inspection was to review the technical issues associated with this event to determine its causes and safety implications. An Augmented Inspection Team was dispatched on September 11, 1987 to conduct the inspection in a timely, thorough, and systematic manner. This reports the results of the inspection.

On September 11, 1987, Oyster Creek Nuclear Generating Station reported to the Nuclear Regulatory Commission that a violation of Safety Limit 2.1.E had occurred in that fewer than two sets of recirculation loop valves were not fully open for a short period of time as required by the limit. This Augmented Inspection Team confirmed by their review and analysis of plant records that this condition existed for about two and a half minutes. The control room operator involved quickly recognized the error and took prompt action to recover from the situation. The team concluded that the event had minor significance from a reactor safety viewpoint for the following reasons:

1. The plant had been shutdown for one day and was being cooled by the shutdown cooling system; this system continued to function throughout the event.

2. With the combination of valves being opened and closed, there was always sufficient fluid communication between the reactor core region and the annulus region of the reactor vessel to assure fluid communication and ascertain fluid level in the core region; this is the reason the Safety Limit was imposed in 1979.

3. Since the imposition of the Safety Limit in 1979, a fuel zone fluid level indicator had been added as part of the enhancement required by the Three Mile Island Action Plan for all nuclear power plants. This level indicator was operable and provided a record of adequate level throughout the event.

The team also inspected the circumstances leading to the event and several of the related activities. A maintenance activity led to a leak in a cooling water system and resulted in an immediate need to secure recirculation pumps; the consequent valving error led to the violation. This maintenance activity, discussed in detail in the report, was conducted with poor adherence to plant procedures, poor communications among the several individuals involved and one apparent lack of skill and knowledge by a licensed reactor operator. The recovery from the leak and the handling of the resulting minor personnel contamination/personal safety problems were accomplished promptly and properly.

Reporting of the event to the Nuclear Regulatory Commission met the four hour reporting requirement for events of this type.

The inspection report details the history of the safety limit, including a brief recap of the May 2, 1979 event that explains its need. Also discussed is the history of the alarm installed in 1986 which notifies the operator when there are an inadequate number of valves full open.

Since a missing or destroyed rec. was identified by the licensee subsequent to the event, some inspection effort was spent examining similar records to ascertain whether there was any pattern of missing records. No pattern was evident from the limited sample examined.

A review of the procedures governing the activities conducted in the control room showed that the activities did not strictly adhere to the procedures written. In this instance, it appears that the recirculation loop operating procedures had not been revised to reflect current plant practices for shutting down and operating recirculation pumps and valves.

2. Background

Oyster Creek is a boiling water reactor located in Ocean County, New Jersey with a power rating of 1930 megawatts thermal and 650 megawatts electric. In May of 1979, as a result of closing all 5 recirculation system valves simultaneously during a transient, a reactor zone water level transient occurred that went undetected until the receipt of the low-low-low level alarm.

Soon after this incident, the NRC imposed a Safety Limit on the facility that required at least two loops to have both the recirculation pump suction and discharge valves open during all plant conditions unless the reactor vessel head was removed. Technical Specification 2.1.E states, "During all modes of operation except when the reactor head is off and the reactor is flooded to a level above the main steam nozzles, at least two (2) recirculation loop suction valves and their associated discharge valves will be in the full open position."

As part of the post-Three Mile Island Action Plan, the NRC imposed requirements for an interlock to prevent less than two loop operation. This was subsequently changed to an alarm in the control room that would actuate if less than two recirculation loop suction and discharge valves were fully open. This alarm was installed during the recent (11R) refueling outage which was completed in December 1986, some 8 years after the original event at Oyster Creek.

On September 10, 1987, Oyster Creek was taken off line at 0107 hours for plant maintenance. Included in this maintenance was a planned job to repack valve V-5-167. This valve is a primary containment isolation valve in the Reactor Building Closed Cooling Water (RBCCW) System. On the September 10 day shift, tagging to support accomplishment of this activity was completed. At 0200 on September 11, 1987, the plant was in cold shutdown, reactor vessel vented, and primary coolant temperature at approximately 140°F. Recirculation pumps B and C were operating; A, D and E were secured and their pump discharge valves closed. The shutdown cooling system attached to Loop E was in operation.

Around 0210 hours on September 11, 1987, during removal of packing from V-5-167, a leak occurred from the packing gland which required securing the RBCCW system. This was done by isolating the drywell portion of the RBCCW system. Since this cools recirculation pump components, the pumps were to be secured. The control room operators, while in the process of shutting down the two running recirculation pumps, entered into a condition which violated the Safety Limit imposed by Technical Specification 2.1.E. The plant remained in this condition for approximately two-and-a-half minutes. The NRC was notified of the event at 0403 on September 11, 1987.

3. Plant Status and Sequence of Events

3.1 Plant Conditions Immediately Prior to Event

The reactor plant was subcritical reading approximately 40 counts per second on the source range instrumentation with the mode switch locked in the shutdown position. Reactor coolant temperature was approximately 141° F with the "C" shutdown cooling loop operating. Reactor water level was being maintained at approximately 160 inches above the top of the active fuel; letdown flow to the reactor water cleanup system was in progress with an auxiliary pump running. Total recirculation flow was approximately 4.3×10^4 gallons per minute (gpm) maintained using the "B" and "C" recirculation pumps. The control rod drive system was in operation with the NC08B pump in service with automatic flow control. The "C" condensate pump was operating and hot well level control was in manual. The (RBCCW) system was operating with both pumps running maintaining a discharge temperature of 89° F. The reactor was vented through the main steam lines. Torus draining was in progress and the "A" and "D" recirculation pump motor generator sets were being warmed up prior to starting. Preparations were being made to conduct maintenance on V-5-167, an RBCCW drywell isolation valve.

3.2 Sequence of Events

During the early morning of September 11, 1987, at approximately 0210, a radiological control technician reported a leak to the control room. The group operating supervisor (GOS) responded to the 23 foot elevation to investigate while the shift personnel prepared to take action in the control room. The GOS recognized the purplish color water as RBCCW system fluid, remembered maintenance was in progress on V-5-167, a RBCCW system containment isolation valve, and reported to the control room that the leak was from V-5-167. After studying piping diagrams, the control room operators (CRO's) prepared to isolate that section of RBCCW system piping, directing equipment operators to manipulate a manual isolation valve while they closed a

motor-operated valve. Action was taken in the control room to remove the recirculation pumps from operation as isolating this section of RBCCW system piping will interrupt cooling flow to the "B" and "C" recirculation pumps. From analyzing available information on plant parameters and alarms, the inspection team's best judgement is that the "B" recirculation discharge valve was shut at approximately 02:17:20 followed immediately by closure of the "C" recirculation discharge valve at approximately 02:17:22. The graphic time history of significant parameters used to draw these conclusions is shown in Figures 3 and 4 and events are described in detail in Table 1. The conclusion that "C" recirculation discharge valve was shut can not be verified, however, the weight of technical evidence supports this conclusion.

As the "B" and "C" discharge isolation valves stroked shut, the alarm annunciator in the control room "Less Than 2 LPS" was actuated. This alarm indicates that less than 2 recirculation system loops are unisolated and is described in the Oyster Creek technical specification as a safety limit. (See paragraph 5.1 for further detail.) At this point with "B" and "C" valves shutting and the annunciator alarming, action was taken by the CRO to immediately open "D" and "A" recirculation discharge valves at approximately 02:17:28 and 02:17:30 respectively. Action could not have been taken to simply reopen the "B" and "C" recirculation discharge isolation valves since the valve closing signal seals in, requiring the valve to full stroke closed prior to permitting the valve to be reopened.

Opening "D" and "A" was accomplished to assure adequate fluid communication between the reactor vessel core region and the annulus region (safety limit basis). This is important as most of the reactor level instrumentation measures water level in the annulus region with the exception of the fuel zone level instrumentation which measures the core region level. (See paragraph 5.2 for further detail.) With the "B" and "C" recirculation pumps still running and the discharge valves shutting, the operator reduced recirculation flow using the master recirculation flow controller prior to tripping "B" and "C" recirculation pumps at approximately 02:18:10 and 02:18:14 respectively. During this time, shift personnel were able to stop the RBCCW leak by isolating RBCCW to the drywell and as a result to the recirculation pumps.

At this point, the recirculation pumps were coasting down and two isolation valves were stroking open and two isolation valves were stroking shut. The less than two loops alarm was still illuminated. The "E" recirculation loop remaind isolated with shutdown cooling system operating throughout the event.

The valve stroke times are such that the "C" valve was able to stroke shut prior to the opening valves clearing the "Less Than 2 LPS" annunciator. The inspection team concluded that when "C" valve closed it was immediately reopened which would support the "C" loop suction temperature decrease identified later in the event. Immediately after "C" loop was reopened, the "Less Than 2 LPS" alarm cleared at 02:19:17, since "A" and "D" valves were now full open. (See Figure 3.)

Subsequently, cleanup letdown flow was stopped and condensate flow was increased to raise reactor water level to 185" to enhance shutdown cooling circulation and develop natural circulation flow. The "B" discharge recirculation valve was opened approximately 27 minutes after start of the event and about 25 minutes prior to the start of the recirculation pumps. V-5-167 was eventually backseated, permitting restoration of RBCCW flow to the recirculation pumps and starting of the "A" and "D" recirculation pumps.

After the initial event and prior to starting recirculation pumps "A" and "D", the loop suction temperatures trended down for the open recirculation loops A, C, and D. This occurred prior to water level being raised to 185" inches above Top of Active Fuel (TAF) to facilitate natural circulation and shutdown cooling flow. Prior to this time, shutdown cooling flow likely was backflowing in the open recirculation loops to cause the suction temperature decline as indicated on Figure 4. Almost 40 minutes after the event, the "A" and "D" recirculation pumps were started causing increased reverse flow through unisolated loop "B" and "C" and increased core flow. The increased flow promoted uniform mixing of the reactor coolant and eliminated any stagnant water. As a result the loop suction temperature profiles responded as indicated in Figure 4. The maximum recirculation suction water temperature occurred at this time and was 155° F.

3.3 Analysis of Safety Significance

The safety significance of this event is considered low based upon the analysis of the events by the inspection team. The basis of the safety limit is to maintain fluid communication between the core region and the annulus region of the reactor vessel. The annulus region is where most of the level instrumentation measures reactor vessel level. Only the fuel zone level instrumentation measures level in the core region. Therefore, to accurately measure vessel level and activate the associated alarms and safety features, at least two loops must be open to facilitate adequate fluid communication between the core region and the annulus, according to Technical

Specifications. The licensee has determined that one loop open would be sufficient to maintain communication and that during a plant shutdown, if no action is taken upon isolating all five recirculation loops, it would take approximately six hours before boiloff of water would lower vessel level to the top of active fuel from the normal water level band. The inspection team review agrees with this analysis.

In this case, two isolation valves were opening as two isolation valves were closing and in addition all five recirculation discharge bypass isolation valves (2" valves) were full open. The inspection team's analysis indicates that the time span from when the first isolation valve started to close until second isolation valve reached the full open position and cleared the less-than-two-loops-isolated alarm, the time was approximately two minutes. The maximum possible time duration for this valve stroke sequence to occur would have been approximately two and one half minutes. Therefore, considering actual valve stroke times and time duration of the sequence, there was no time period during which the core region was completely isolated from the annulus region. In addition the pump discharge bypass valves were open throughout.

In conclusion, this event is considered to be a literal violation of the safety limit Technical Specification, but its safety significance is considered low. The basis for this conclusion is that core cooling was always adequate due to the low decay heat levels, fluid communication between vessel regions was always maintained and, in addition, the fuel zone level instrumentation was functional and indicated that the fuel region water level was always maintained at appropriate levels for a shutdown reactor.

4. Maintenance Activities Leading Up to the Event

A leak of approximately 250 gallons of mildly radioactive liquid from the Reactor Building Closed Cooling Water (RBCCW) system occurred while replacing the packing on isolation valve V-5-167 shortly after 0200 hours on September 11, 1987. This occurred over a period of 10-15 minutes, contaminated the worker involved in the valve packing, and required plant operations to secure the leak and render assistance to the worker.

The RBCCW system is a closed loop cooling water system which provides cooling to the following components in the drywell:

-- Reactor Recirculation
pump and motor
coolers (5)

-- Drywell cooling
units (5 with one spare)

-- Drywell equipment
drain tank (1)

The RBCCW system normally operates at approximately 100 psi, with temperature maintained in the low end of the 70° to 150° F operating band. V-5-167 is an outboard primary containment isolation valve used to isolate the RBCCW system drywell cooling return header. V-5-167 is a six inch motor operated gate valve mounted upside down (Fig. 1) in that the gate is above the valve stem (packing leakage travels down the valve stem).

V-5-167 failed to meet its required valve stroke time during a surveillance test conducted on July 31, 1987. An investigation determined that a corrosion inhibitor (Boron Nitrate) used in the RBCCW system had leaked from the packing down the valve stem (valve is mounted upside down). This packing leakage allowed boron crystals to form in the limitorque operator stem nut, increasing valve stem friction to the point that the limitorque motor cut off on high torque. The licensee cleaned up the stem nut and applied a nuclear grade lubricant on the valve stem while stroking the valve. This action enabled the valve to meet its required stroke time demonstrated during MOVATs testing conducted on August 3, 1987. However, the MOVATs testing also determined that the motor operator was still experiencing higher than desired motor current during valve operation. As the long term corrective action, the licensee initiated a work order on August 6, 1987 to repack the valve with Chesterton live loaded packing. This packing usually prevents or minimizes packing leaks and would prevent recurrence of the previous problem. Until the valve could be repacked, the licensee implemented additional action to ensure valve operability. This action included partially closing the valve monthly using the manual operator and fully closing the valve quarterly using the electric motor operator.

The maintenance planner specified the use of a Chesterton repacking procedure (700.1.030) which required the valve to be isolated and vented as a prerequisite to repacking the valve. On the morning of September 10, 1987, the Maintenance Supervisor (MS) submitted a tagging request to the Group Shift Supervisor (GSS) which was in accordance with the procedure referenced in the work order.

The GSS on shift then correctly advised the MS that the valve could not be isolated at the time since the system was supplying cooling to critical loads which could not be secured, and that the tagging request would have to be revised to repack the valve on its backseat.

The MS informed his superiors that he did not think he could perform the repacking work under the procedure referred to in the work plan. His management discussed the adequacy of the procedure with plant engineering (including the procedure's author) and the maintenance planner. A communication misunderstanding occurred in that plant engineering's position was that the procedure would be adequate only with a temporary change to the

procedure or additional precautions. However, maintenance management and the maintenance planner concluded from these conversations that plant engineering concurred that the referenced procedure was adequate without any additions or modifications (communication problem #1). This is contrary to procedure control procedure 107 which stipulates a modification to a procedure prerequisite requires a temporary change to the procedure with approval of one of the following personnel:

1. Plant Operations Director
2. Manager Plant Operation
3. Operations Control Manager
4. Manager Plant Materiel
5. Plant Engineering Director
6. Director/Deputy Manager Radiological Controls (Rad Con requirements only)
7. PRG Chairman or Vice Chairman
8. Director/Deputy Director, Oyster Creek

During the day (September 10), maintenance management held additional discussions within their department including the maintenance functional manager and concluded the valve could be repacked on its backseat without procedure modification. In the afternoon, the 8 A.M. to 4 P.M. shift MS submitted a revised tagging request that specified the valve was to be placed on its backseat and tagged for repacking. The MS told the GSS he desired the valve to be manually placed on its backseat and a tag hung on the valve. The GSS refused to manually backseat the valve due to concern for damaging the valve stem and backseat and gave instructions to backseal electrically. The GSS incorrectly thought that he was backseating the valve by giving the valve a two second additional open signal on the normal control switch. The GSS believed that the two second additional open command would allow the torque switch to adequately backseat the valve without any valve damage (technical problem #1). This method to backseat the valve is not in accordance with any station approved procedure (procedure problem #1). This action, accomplished late in the 8:00 a.m. to 4:00 p.m. shift, did not place the valve on its backseat as the limit switch is not bypassed in the open direction. The GSS assumed the valve was being repacked due to a visible packing leak. He reasoned that if his attempt to backseat the valve was unsuccessful, the packing leak would still be evident. The GSS cautioned the MS to look for and report any indication of packing leakage so an operator could be dispatched to manually backseat the valve. The station has two approved procedures which could have been utilized to backseat this valve, a manual backseating procedure (A100-SMM-3917.06) and an electrical backseating procedure (700.2.012), both of which require completion of a data sheet which was not filled out. The electrical backseating procedure (700.2.012) is not applicable for the subject valve and could only have been utilized by first obtaining concurrence from plant engineering, which was not done.

A Reportable Occurrence (50-219/81-65/01T) occurred on December 23, 1981 and involved limitorque operators and isolation valves for the Isolation Condenser System which were found to have defects which did affect the operability of one valve and could affect those on other systems that perform a reactor coolant pressure boundary and primary containment isolation function. Corrective action taken for this incident required that all station operating procedures would be reviewed and modified such that safety system isolation valves would be backseated in an approved manner. Neither the GSS, the maintenance planner, nor the MS knew that approved station procedures were available on the proper method to backseat this valve (procedure problem #2). A detailed discussion of operator training in the area of limitorque valve operations is discussed in paragraph 6.3 of this report.

A second communication misunderstanding occurred when the dayshift GSS advised the MS that the valve was very close to its backseat or just on its backseat. The MS left the control room understanding that the valve had been electrically backseated (communication problem #2). The valve was in fact, approximately one and 1/4 turn off its backseat. The dayshift GSS directed that the breaker that provides power to V-5-167 motor operator be tagged out. This is the only tag that was hung for the repacking of V-5-167. This is contrary to equipment control procedure 108, paragraph 5.1.9, which requires that "If equipment or piping is to be opened, valves and switches shall be aligned and tagged so as to insure that the work does not present a hazard to personnel or equipment from pressure, vacuum, fluids, gasses, or radioactive contamination..." Also, paragraph 5.1.15 requires that "If a tag is placed on a component's power supply, a tag shall also be placed on each remote control. A tag need not necessarily be placed on the component's manual operator in this case, if the manual operator or its associated component is not part of the safety boundary."

The 4 to 12 P.M. shift on September 10 was unable to accomplish the repacking due to required machining on the packing follower. The September 11, 12:00 midnight to 8:00 A.M. operation shift was turned over with the information that the valve was backseated (communication problem #3). The mechanic performing the maintenance went to the control room to verify valve position prior to commencing work. The operating shift confirmed that the valve was backseated. The mechanic proceeded to remove the valve packing one ring at a time and observed no leakage while removing the first four rings of packing. At approximately 2:10 A.M., the mechanic commenced removal of the fifth ring of packing (8 rings total) when the remaining packing blew out causing a sizable RBCCW system leak. Personnel in the area responded promptly and took effective action to isolate the leak and minimized the size of the spill area. The radiological impact of this event is discussed in paragraph 7 of this report.

Summary of Maintenance Activities

1. The final work package as written effectively communicated to the MS and the GSS that V-5-157 was required to be on its backseat to perform the valve repacking. The GSS overruled the MS's request that the valve be manually backseated and tagged. The GSS was not aware of the approved station procedures for backseating this valve. The GSS did not request guidance from plant engineering or Technical Functions as to the method or procedure which should be used to backseat this valve.

   The GSS developed and implemented his own method which he thought would place the valve just on its backseat or very close to its backseat for maintenance work that would remove all the valve packing from the stuffing box. The GSS and MS did not understand from the work package that all the packing would be removed from the stuffing box. The GSS's method actually placed the valve approximately one and a quarter turns off its backseat.

2. The system conditions for the repacking maintenance did not meet the prerequisites in the procedure referenced in the work package. Procedure 107 requires a modification to a procedure prerequisite obtain approval of senior level review, which was not done.

   The work package did not specify the specific procedure to be utilized to backseat the valve for the maintenance work (plant-wide standard practice).

3. The equipment control procedure 108, Sections 5.1.9 and 5.1.15 were not complied with in that the manual operator on the valve was not tagged and its operation affected the safety boundary. Several other procedure problems, three communication problems, and a technical problem were identified and are highlighted in the above write-up.

5. Safety Limit

5.1 History of Technical Specification Safety Limit 2.1.E

On May 2, 1979, an event occurred at Oyster Creek involving closure of all five recirculation pump discharge valves shortly after a reactor scram and subsequent initiation of the isolation condensers. Because the five discharge valves were closed, there was a break in the communication of fluid between the annulus and reactor fuel spaces within the reactor vessel. Since the reactor vessel low and low-low level alarms are sensed from level indicators that monitor water level in the annulus region, the loss of water inventory from the fuel region caused by isolation condenser operation remained undetected until the low-low-low level alarm was received. This alarm was received because its water level system's variable leg is the core spray sparger which is within the reactor fuel space region of the vessel.

Prior to this event, the Technical Specifications (TS) did not specifically address any requirements for recirculation loop valve positions. The TS did, however, specify in Safety Limit 2.1.D, the minimum water level during the shutdown mode. Subsequent to this 1979 event, TS Amendment No. 36 was issued on May 30, 1979 to amend Safety Limit 2.1.D to specify the minimum water level in all modes of operation and add Safety Limit 2.1.E that required two recirculation loops to remain open during all modes of operation except with the reactor vessel head removed. At about the time this amendment was issued, all affected site procedures were revised to reflect the new Safety Limit and plant operator training was updated.

5.2 History of the Recirculation Loop Alarm/Interlock

As a result of the incident described in Paragraph 5.1 above, the licensee investigated the use of an interlock scheme to prevent isolating more than three recirculation loops.[1]* Following the Three Mile Island (TMI) 2 event, the NRC issued NUREG-0626 on May 7, 1980. Recommendation A.8 stated interlocks should be installed on all nuclear power plants without jet pumps for recirculation to assure that at least two recirculation loops are open for recirculation flow for modes other than cold shutdown. Later in 1980, position II.K.3.19 in NUREG-0660 and 0737 reiterated this statement and clarified it to be a post TMI Action Item. The licensee's initial responses to these NUREGs indicated they intended to install the interlock.[2] An installation specification was developed, and the project planned for the 1983-84 refueling outage. It was subsequently cancelled. Because of delays encountered with designing and installing the modification, the licensee took compensatory measures by adding hinged covers to the 3F control room panel that covered the control switches for each pair of recirculation loop isolation valves. An engraved caution sign was affixed to each cover alerting the operators to avoid less than three loop operation.

On March 14, 1983, the NRC issued a Confirmatory Order to GPUN setting forth dates by which various TMI Action Plan Items were to be completed. This Order stated that the licensee had until the end of the 11R outage to complete installation of the interlock modification. In September 1985, the licensee informed NRC by letter[3] that an alarm would meet the functional requirements of an interlock. The letter went on to say the alarm provides positive active indication to the operator that a fourth loop has been isolated and further explained that the NRC Staff evaluation, presented in NUREG-0626, did not take into consideration a fuel zone level monitoring system for Oyster Creek vintage plants. During the 1979-80 Cycle 9 refueling outage wide range fuel zone level indication and a recorder were installed. With recirculation pumps tripped, this instrumentation provides the reactor operator with level indication in the core region.

*Footnote references can be found in Appendix A.

A meeting between GPUN and NRC/NRR to discuss this proposal occurred on October 9, 1985. The NRC requested further amplification of the proposal which was provided by GPUN in a January 1986 letter.[4] This letter explained that: (1) the alarm on closure of the fourth loop isolation valve would alert the operator that a Safety Limit had been violated, (2) analysis concluded only one unisolated/recirculation loop is required to provide coolant communication between the annulus and core regions, (3) with one loop open in addition to the five 2-inch discharge valve bypass valves open the recirculation flowrate is about 5 to 6 times the boiloff rate in the core region, thus, indicating the conservativeness of the Safety Limit that requires two open loops, (4) with the alarm modification, an operator would have to disregard training, violate procedures and ignore posted warnings, and be unaware of the significance of the switch covers in order to violate the safety limit.

NRC accepted GPUN's proposal for the alternative in a July 1986 letter[5] that accompanied TS Amendment No. 106. This letter stated it documented revising the requirement in the March 14, 1983 Order to agree with the change from an interlock to an alarm. NRC concluded that the alarm only modification and trained operators would have the same effect as the interlock without the complexity introduced by the interlock. The alarm was installed during the 11R outage and has been functional during cycle 11 operation. It is a unique alarm in that when it alarms the alarm light has a green background whereas other alarms have a white background. The engraving on the alarm states, "Less than two recirc loops open". It should be noted that, until the recent installation of this alarm, operators would not have been alerted had they violated Safety Limit 2.1.E.

6. Human Factors in the Control Room

6.1 Procedural Requirements For Operation of Recirculation Loop Valves

Station Procedure 301, Nuclear Steam Supply System, contains the procedure for operation of the recirculation loop valves. Paragraph 4.2.2, which is a precaution, restates Safety Limit 2.1.E.

Paragraphs 4.3.3 and 4.3.4 are the steps to open or close a suction or discharge valve and appear to be poorly written regarding closing the valves as described in the last paragraph of this section.

The steps read:

"4.3.3 To open suction or discharge valves, hold their respective control switches on 3F in the "OPEN" position and then release the switch.

4.3.4 Repeat the same procedure for valve closure.

NOTE: Torque switches shut off the valve drive motors to prevent strain on valve parts should an obstruction or damaged valve prevent proper operation. If the valve does not stroke in its normal time ± 10 sec. (suction valve - 2 min., 20 sec.; discharge valve - 2 min.), release the control switch and investigate the cause of the malfunction."

The last sentence of the "NOTE" in 4.3.4 implies the operator holds the switch in the closed position for a specified time. Yet the initial action in 4.3.4 is to repeat 4.3.3 which requires the operator to position the switch and then release it. Because the 'close' contact locks in after the switch is positioned to close the valve, it is not necessary for the operator to hold the switch in the close position and, in actual practice, the operators do not hold the switch.

6.2 Procedural Requirements for Operation of the Recirculation Pumps

Station Procedure 301 also contains the steps for normal operation of the recirculation pumps. Steps 6.2.4 and 6.3.1 discuss how to change recirculation flow and require that the individual recirc pump speed control units should be in AUTO. In actual practice, operators do not adhere to this requirement. The individual speed control units are left in BALANCE. While this is not a problem from the operational performance, it is a departure from present practice.

Paragraph 7.0 of Standard Procedure 301 contains the steps required to remove a recirculation pump from service under normal conditions. It restates Safety Limit 2.1.E requirements. The sequence is designed for normal plant operation, not an emergency type situation. The steps require the operator to, in order:

(1) Run the speed for the pump to be secured to the minimum possible value.

(2) Check open the pump discharge bypass valve and then close the discharge valve. There is a CAUTION in the procedure at this point that states: The suction and main discharge valves of at least two (2) recirculation loops must remain open.

(3) Stop the recirculation pump drive motor.

This procedure is lacking in specifics if three pumps are secured and their respective discharge valve closed which is generally the case when the plant is in cold shutdown. The steps create the potential during cold shutdown that the operator would unknowingly violate safety limit 2.1.E; that is to say three valves are shut during cold shutdown but the procedure tells the operator to close the valve associated with the pump to be secured thereby resulting in four of five valves closed. The quandary could be avoided if the procedure specifically instructed the operator to open an idle loop discharge valve before entering the shutdown sequence for the fourth pump.

Abnormal Procedure 2000-ABN-3200.19, RBCCW Failure Response, contains the steps required to remove recirculation pumps under abnormal conditions, e.g., complete loss of RBCCW flow, major RBCCW leak, etc. The steps in paragraph 3.1 of this procedure require the operator in sequence to:

(1) Scram the reactor;

(2) Trip all operating recirculation pumps; and

(3) Confirm that all recirculation pump suction and discharge valves are open.

Based on a post event review of available data, it appears the operator did not respond in accordance with this procedure.

6.3 Operator Training - Motor Operated Valves

A review of operator training in the area of motor operated valves (MOVs) indicated that all licensed operators received training on MOVs in December 1986, January, and February 1987. The training described, in part, that the motor can be stopped by a signal to its control circuit from either a torque switch or a limit switch. It was explained that valve travel in the closed direction can be stopped either by limit switch or torque switch and in the open direction by limit switch. Additionally, a torque switch will stop valve travel in either direction to prevent valve damage. This would prevent, for example, damage to the valve if the open limit switch failed.

The training material consisted of several handouts, a video tape presentation, and an actual valve and motor operator. The classroom material that was discussed included backseating for leakage control, including electrical backseating using site procedures and standing orders. It appeared from a review of this training that the recipients should have been aware that a MOV could not be backseated electrically using the valve's control switch. This knowledge should have been reinforced by the routine use of site procedures to electrically backseat valves. These procedures require overriding a contactor locally at the breaker and monitoring current to the motor using an ammeter.

7. Contamination Incident

The maintenance worker performing the valve repacking was sprayed with water from the RBCCW system. He was quickly removed from the area and his clothing removed. He was frisked and found to have less than 100 cpm contamination on his skin. He then showered and had a whole body count performed. The results of the whole body count identified no abnormal activity. The individual also received water containing boron nitrate in his eyes and was treated for eye irritation.

One AIT inspector discussed the incident with station personnel, reviewed the applicable Radiation Work Permit (RWP), the results of a whole body count on the affected individual and a statement by a licensee physician who examined the individual. During the review, the inspector noted that the individual performing the maintenance was dressed only in rubber gloves and a hood. Review of the RWP identified that anticontamination clothing was only required if the surface area surrounding the valve was contaminated. The inspector asked about this requirement. Licensee representatives stated that based on conditions in the RBCCW system, the backseating of the affected valve and the fact that a catch basin was being placed under the work area to catch minor leakage, no further protection was required. The licensee representative further stated that had they expected a large amount of water, plastic suits would have been required. The inspector, in consultation with Region I radiation protection personnel, determined that the licensee's actions were appropriate for the circumstances. The inspector had no further questions regarding this matter.

8. Notification

On September 11, 1987, at about 0218, while performing maintenance on the Reactor Building Closed Cooling System (RBCCW) leakage occurred from the system and an operator proceeded to shutdown the two recirculation pumps, which were in service at the time as a result of a need to isolate the

RBCCW system leak. This is in violation of Technical Specification 2.1.C, Fuel Cladding Integrity, which requires two recirculation loops to have their suction and discharge valves in the full open position during all modes of operation. At the time of the event, the Oyster Creek plant was in cold shutdown conditions. The licensee notified the NRC Operations Center in Bethesda, MD of this event at 0403 on September 11, 1987 (one hour and forty-five minutes after the event).

10 CFR 50.72(b)(2) requires that for such an event that occurred at Oyster Creek, the licensee shall notify the NRC as soon as practical and in all cases within four hours of the occurrence.

The staff is presently reviewing the licensee's internal management notification of this event. Preliminary information indicates that operations management was not advised of the event in a timely manner and that some details were not communicated to management by the operations staff.

9. Review of Sequence of Alarms Recorder Tapes and Control Room Operator Logs

Late on the morning of September 11, 1987, the licensee reported to NRC that a portion of the Sequence of Alarms Recorder printout was missing. Later in the day, an operator disclosed that he tore off the tape, discarded some in wastebaskets and threw some in a toilet. The circumstances surrounding this missing record are subjects of investigation by the licensee and by NRC. The inspection team examined a sample of similar records to ascertain patterns.

A review of the sequence alarms recorder tapes from the period of August 28, 1987 at 3:15 A.M. through September 10, 1987 at 4:53:00 P.M. (two large rolls of recorder tape) did not reveal any evidence that recorder tape had been destroyed or missing. The inspector also reviewed Control Room Operator logs for the period of August 1, 1987 through September 10, 1987. Based on his review of this information, the inspector concluded that control room operator logs were in order.

In Inspection Report 50-219/86-38, the resident inspector of Oyster Creek stated in his review of an event which occurred on December 27, 1986, that, "During the course of reviewing this sequence of events, the inspectors' investigation was hampered by the sequence of alarm recorder missing approximately seven hours of information. Apparently, the operator neglected to insert new paper tape when the recorder ran out." Based on the fact that similar information was missing during the Safety Violation event, the inspector again reviewed the information provided on the subject tape. The inspector concluded that there were no irregularities on the subject sequence of alarms recorder tape. The physical examination of the tape corroborated the original view that the operator neglected to insert a new tape when the recorder tape ran out.

10. Licensee Review of Event

By the time the AIT leader arrived at the site on September 11, 1987, the licensee had formed two internal task groups and engaged investigative consultants. One onsite review group examined the maintenance activities discussed in Section 4 of this report. The onsite group completed its review during this inspection period and is formulating corrective actions. Inspector interaction with the licensee review found it to be objective and self-critical in its determination of root causes. The second effort was the technical review and analysis of the event and its safety implications. While considerable work is still in progress, inspector interactions with members of this group found the effort to be directed to complete consideration of the event and its ramifications and to be a detailed reconstruction of the sequence of events and explanation of exactly what happened. The results of this group's work will be focused toward submitting the required (TS 6.7) report to the Commission.

The investigative consultants were retained by the licensee to examine the circumstances of missing records. The investigation was ongoing at the conclusion of this inspection and was beyond the scope of the AIT charter.

11. Conclusions

On the basis of our review and analysis of plant records as discussed in this report, the inspection team has concluded that a violation of Safety Limit 2.1.E had occurred in that fewer than two sets of valves were fully open.

The inspection team also concluded that the event has minor significance from a reactor safety viewpoint for the following reasons:

1. The plant had been shut down for one day and was being cooled by the shutdown cooling system as described in Section 4 of this report.

2. With the combination of valves being opened and closed, the augmented inspection team has concluded as discussed in Section 3 of this report and depicted in Figure 3 that there was always sufficient fluid communication between the reactor core region and the annulus region of the reactor vessel to ascertain fluid level in the core region.

3. The fuel zone level indicator was operable throughout the event and, as discussed in Section 3 of this report, the water level was always capable of being monitored.

As discussed in Sections 3 and 4 of this report, the Augmented Inspection Team determined that a maintenance activity led to a leak in a cooling water system which resulted in a rapid need to secure recirculation pumps and a consequent valving error led to the violation of the safety limit. On the basis of our review of maintenance activities, and as discussed in Section 4 of this report, the inspection team concluded that:

1. The final work package as written effectively communicated to the MS and GSS that V-5-167 was required to be on its backseat to perform the valve repacking, but the GSS overruled the MS request that the valve be manually backseated and tagged. The GSS was not aware of the approved station procedure for backseating the valve and did not request guidance from Plant Engineering nor Technical Functions as to the method or procedure which should be used to backseat the valve.

2. The system conditions for the repacking maintenance did not meet the prerequisites in procedure 107 referenced in the work package.

3. The equipment control procedure 108, sections 5.1.9 and 5.1.65 were not complied with in that the manual operator on the valve was not tagged.

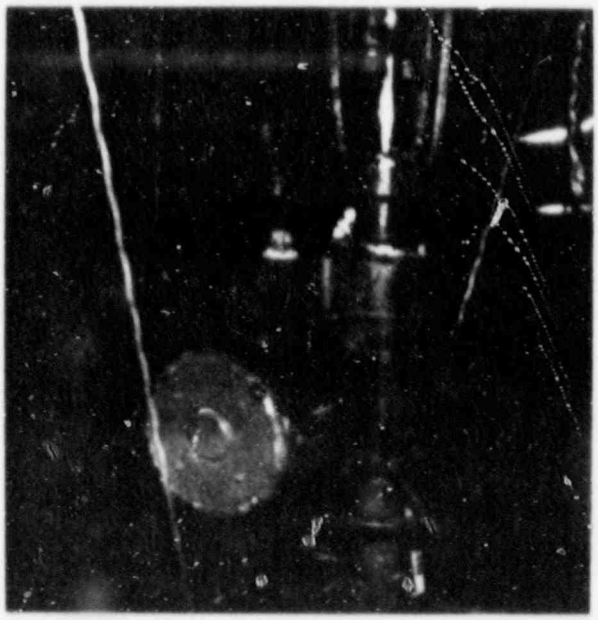
Based on our review of procedures governing the activities conducted in the control room, the inspection team concluded that the activities did not strictly follow the procedures written. In this instance, it appears that the procedures had not been revised to reflect existing plant practices for shutting down and operating recirculation pumps and valves. Details of our review are presented in Section 6 of this report. Further, all licensed operators received training on motor-operated valves (MOV) in the period December 1986 - February 1987. From lesson plans, operators should have been aware that MOVs could not be backseated electrically using only the control switch.

The inspection team also reviewed the recovery from the leak and the handling of the resulting minor contamination/personnel safety problems. As discussed in Section 7 the team concluded that actions taken were accomplished promptly and properly and were appropriate for the circumstances.

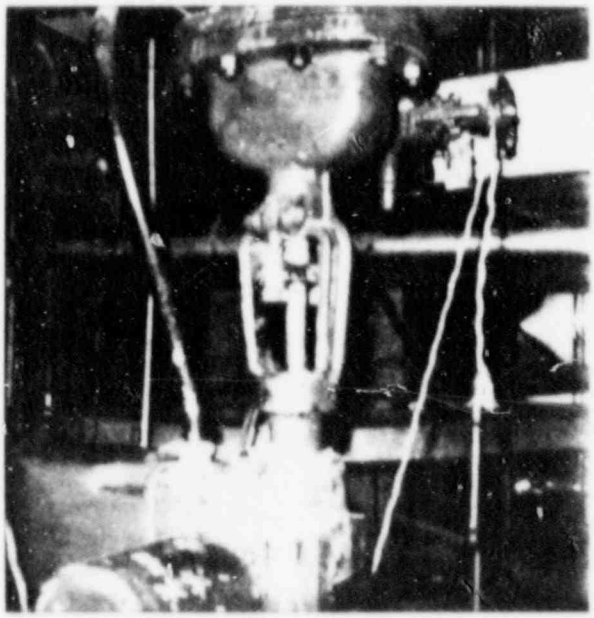
The Nuclear Regulatory Commission was notified approximately one hour and forty five minutes after the event. Therefore, the licensee met the four hour reporting requirement of Section 50.72, paragraph (b) of 10 CFR 50.

The operators' actions subsequent to the event were not reviewed during this inspection. The reviews of those activities and the corrective actions taken as a result of licensee evaluation will be conducted separately.

FIGURE 1

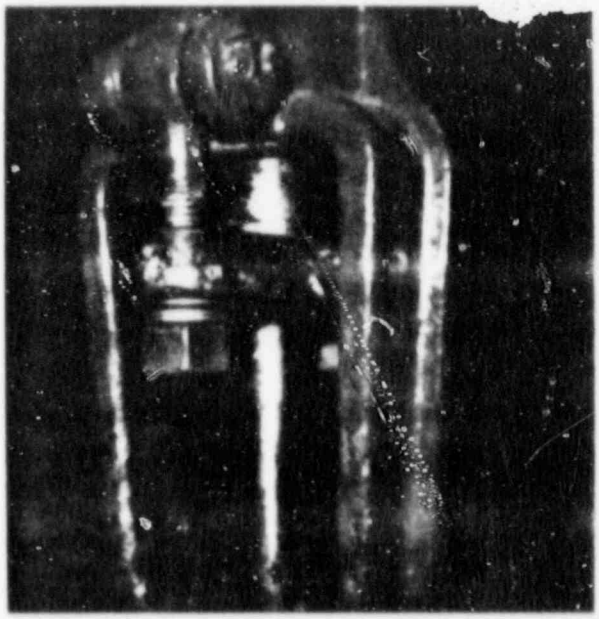


VALVE OPERATOR AND HANDWHEEL



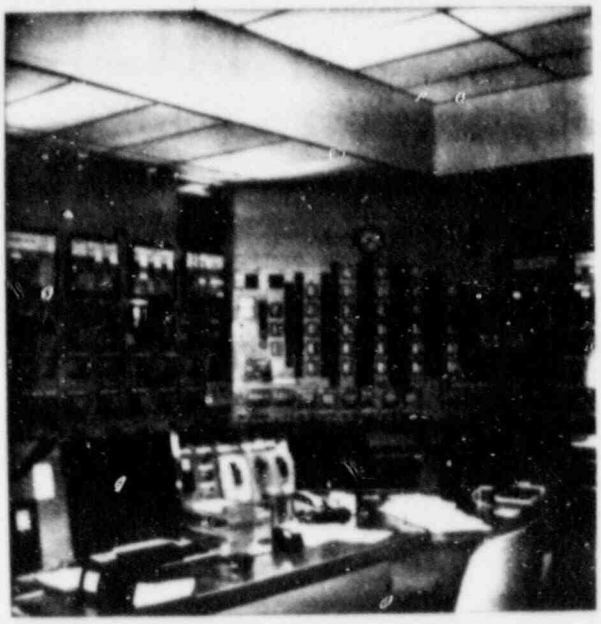
BONNET AND STEM AREA

VALVE V-5-167

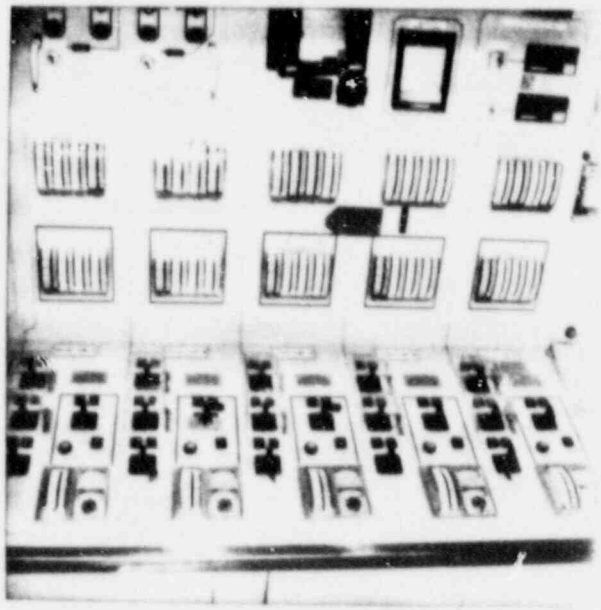


STEM PACKING NUT AREA

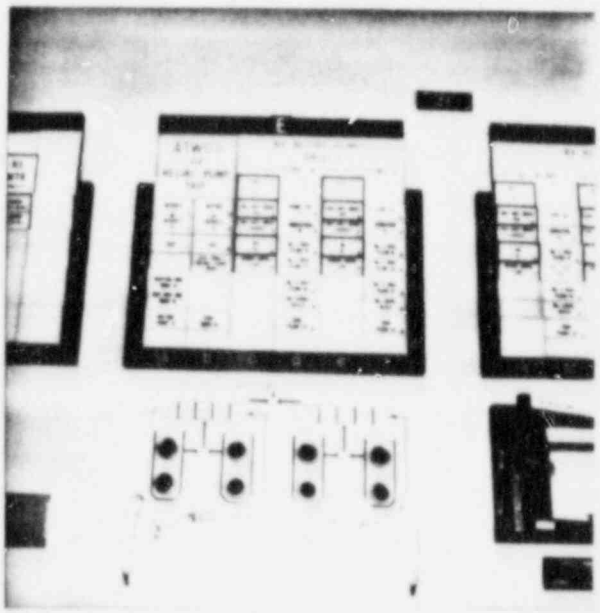
FIGURE 2



View of 3F and 4F panels from behind control room desk.



View of 3F panel. Five individual recirc pump speed controllers on bottom. Second and third row of swtiches are for recirc suction and discharge valve operation.

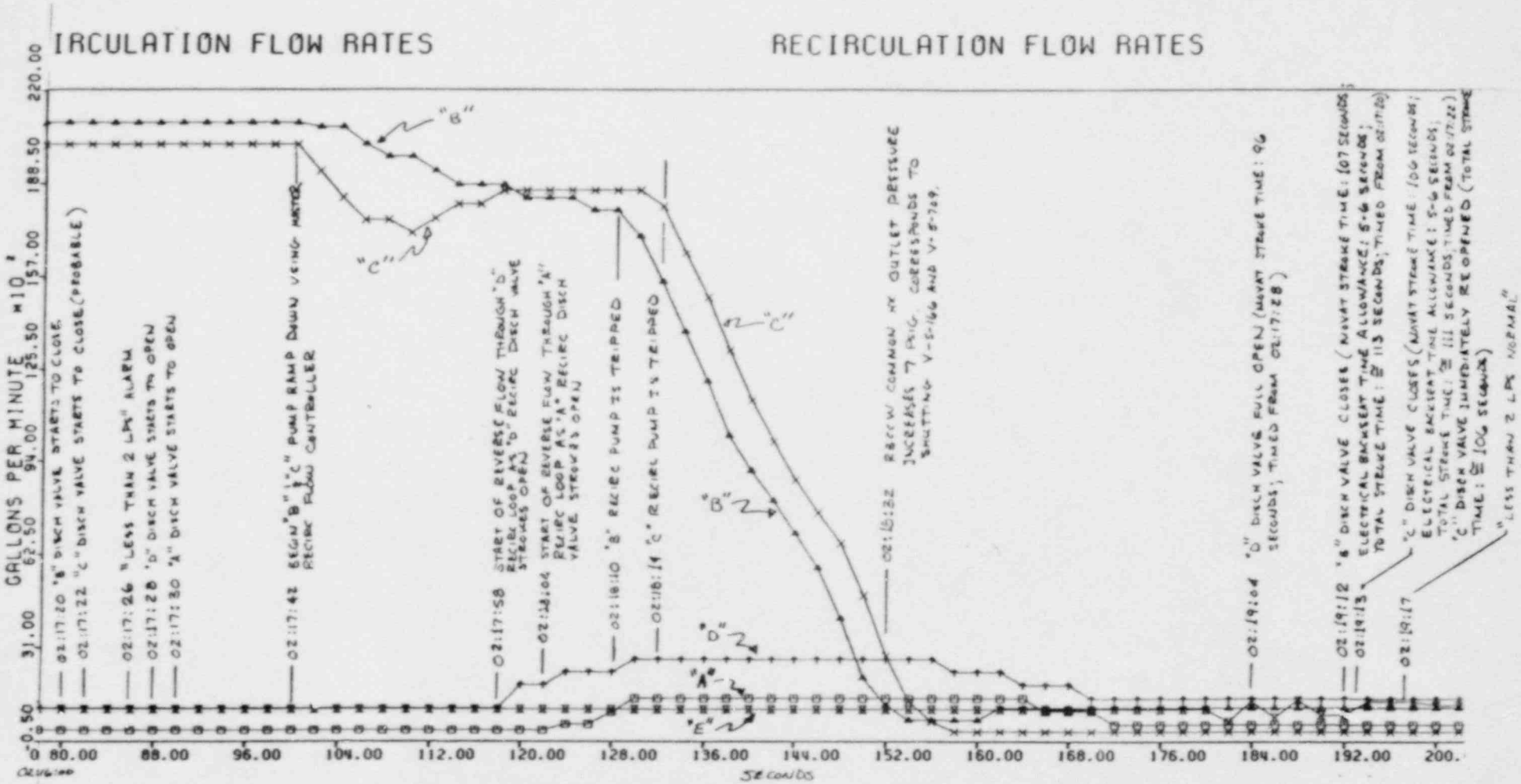


Alarm display E on 3F panel. Alarm 4-b reads "LESS THAN 2 RECIRC LOOPS OPEN."

Seouence of Alarms Recorder (SAR) with tape coming out.

FIGURE 3

PLOT OF PROCESS COMPUTER DATA



	POINT ID	E.U.
⊡	FTIA60A1	GPM
▲	FTIA60B1	GPM
×	FTIA60C1	GPM
+	FTIA60D1	GPM
⧖	FTIA60E1	GPM

	POINT ID	E.U.
⊡	FTIA60A1	GPM
▲	FTIA60B1	GPM
×	FTIA60C1	GPM
+	FTIA60D1	GPM
⧖	FTIA60E1	GPM

FIGURE 3.

ECONDS 02:16:00.800 09/11/1987 → 02:26:00.500 09/11/1987

RATE 2. SECONDS 02:16:00.800 09/11/1987 → 02:26:00.500 09/11/1987

FIGURE 4

PLOT OF PROCESS COMPUTER DATA

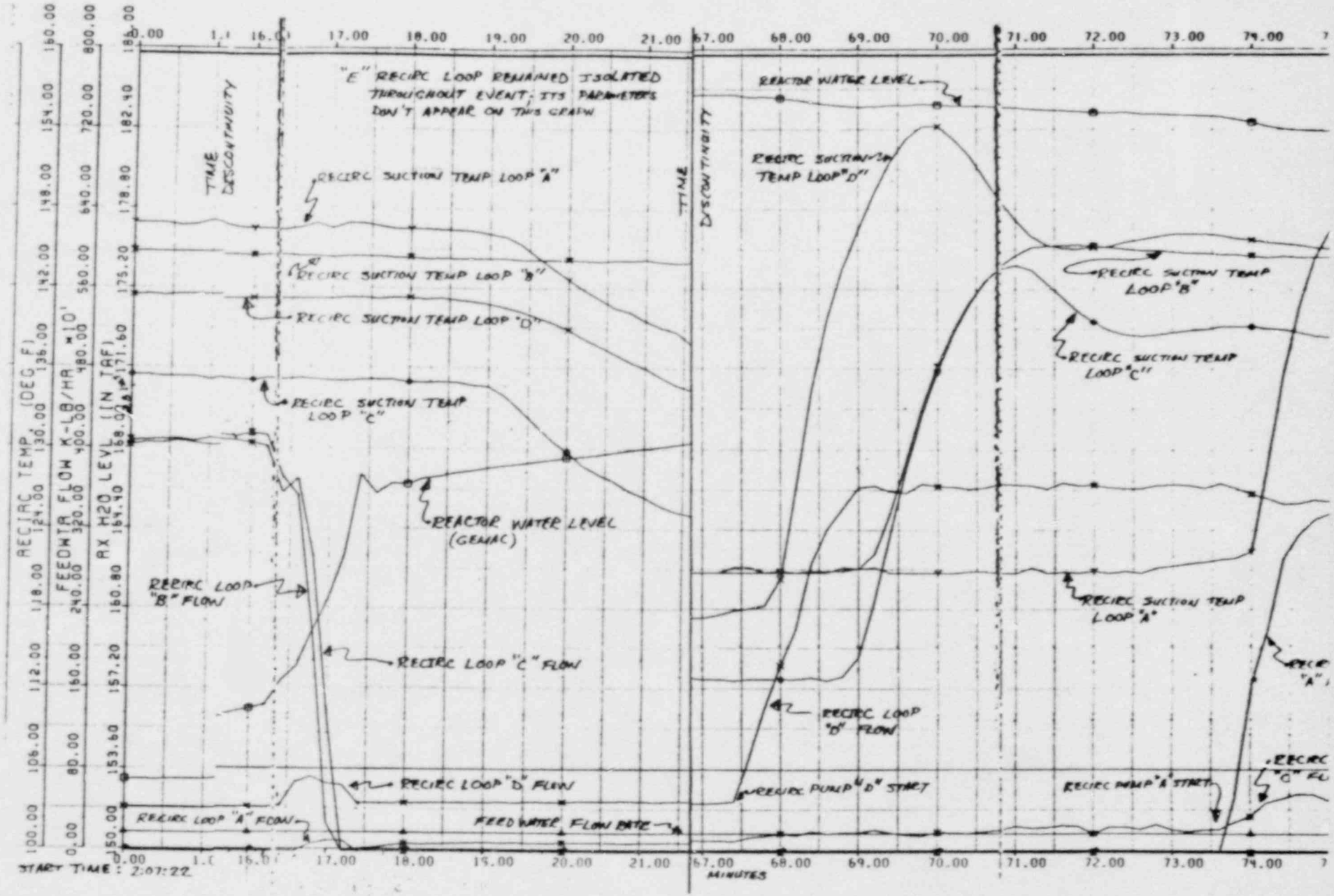


TABLE 1

SEQUENCE OF EVENTS
OYSTER CREEK EVENT, SEPTEMBER 11, 1987

Time	Source	Event
02:02	Control Room Recorder	"A" and "D" motor generator set motor winding temperatures start to increase. "A" and "D" recirculation pumps are not operating. Temperatures increase from approximately 85° F to 100° F over a period of about 16 minutes. The recorder chart shows these temperatures remained at 100° F for about 25 minutes until "A" and "D" recirc pumps were started.
02:10 Estimated Time*	Operator Interviews, Calculations, Security Computer	Major spill reported to control room from 23' elevation by radiological control technician
		Group Operating Supervisor (GOS) dispatched to 23' elevation
		GOS reports water coming from RBCCW
		Shift personnel suspect leak is coming from V-5-167 which was undergoing maintenance
		Leak confirmed to be from V-5-167
2:17:20	SAR & Analysis	"B" recirculation discharge valve started to close. This time is based on the operator closing "B" recirculation discharge valve shortly before the annunciator alarm "less than 2 LPS" actuating. Time was allowed for "B" recirc discharge valve travel from an electrically backseated position. (From backseat to 95% limit switch position: 5-6 seconds; valve stroke time: 107 seconds.)

Time	Source	Event
02:17:22	SAR & Analysis	Probable "C" recirculation discharge valve closure commences. This time as above is based on considered operator action prior to actuating the annunciator indicating less than 2 loops are operating or idle. (Backseat to 95% limit: 5-6 seconds; valve stroke time: 106 seconds.) In addition, examining the "B" and "C" flow coastdown (see Figure 3), no gradual tailoff is evident as would be apparent with the discharge valve full open. Comparing the flow coastdown for "B" and "C", the flow decrease compares favorably. Therefore, whatever actions taken to produce the "B" flow curve would have to be duplicated for the "C" loop. Note that the operator opened "A" and "D" recirculation valves within 2 seconds.
02:17:26	SAR & Analysis	"Less Than 2 LPS" annunciator actuated in control room and printed on the sequence of alarm recorder (SAR). The actual time of the annunciator could not be determined exactly from the SAR paper tape but a time check on the tape immediately after the "Less than 2 LPS" alarm indicated 02:17:45. This time was corrected to the new plant computer system time to yield 02:17:38. This time represents the latest time the alarm could have actuated. Based on the valve stroke times to clear the annunciator and analysis of plant parameters 02:17:26 was considered a probable time for annunciator actuation.
02:17:28	SAR & Analysis	"D" recirculation discharge valve starts to open. This time was estimated from the "Less than 2 LPS Normal" indicated on the SAR tape considering valve stroke time and analysis of plant parameters. (Valve stroke time: 96 seconds)
02:17:30	SAR & Analysis	"A" recirculation discharge valve starts to open. This time was estimated from the "less than 2 LPS normal" indicated on the SAR tape considering valve stroke time and analysis of plant parameters. (Valve stroke time: 107 seconds)

Table 1

Time	Source	Event
02:17:42	Analysis	"B" and "C" recirculation pumps are run back in automatic control from the master recirculation flow controller. The computer graphical representation of recirculation flow for the 9/11/87 event almost exactly duplicates a 9/10/87 recirculation flow response in which both "B" and "C" recirc pumps were ramped down in automatic control. The instrumentation and control supervisor had installed strip chart recorders monitoring "B" and "C" recirculation pump controller responses for a troubleshooting effort to correct flow oscillations with the "C" recirculation pump. The 9/10 controller signal matches the 9/11 recorded controller response for "B" recirculation pump during the event. "C" recirculation pump controller was not available for comparison as the strip chart recorder had run out of paper at approximately 1800 on 9/11/87. Comparing controller signal responses and recirculation flow responses for 9/10 and 9/11, it was concluded that the responses were essentially the same and that the operator had ramped both "B" and "C" recirculation pumps back together using automatic control from the master recirculation flow controller.
02:17:58	Analysis	Reverse flow is indicated in "D" recirculation loop as "D" recirculation discharge valve strokes open.
02:18:04	Analysis	Reverse flow is indicated in "A" recirculation loop as "A" recirculation discharge valve strokes open.
02:18:08	SAR & Analysis	RBCCW isolation alarm (due to closing V-5-166) V-5-166 and V-5-709 were closed to stop RBCCW leak
02:18:10	Analysis	"B" recirculation pump tripped. "B" recirc loop flow starts to coast down "B" motor generator (MG) set motor winding temperatures start decreasing from ≈ 137° F to about 90° F over a period of hours. "B" pump motor winding temperatures decrease slightly. (Control room recorder)

Time	Source	Event
		The strip chart recorder that was monitoring "B" recirculation pump controller signals at the time of the event captures the "B" recirculation pump trip. The blind controller output signal indicates a step decrease in demanded recirculation flow, followed immediately by the pump trip indicated by the tachometer feed back signal dropping precipitously to zero and the blind controller output signal immediately positioning to the low limiter setpoint which on the chart is reflected as approximately 38 ma. (An I & C supervisor later recalibrated the recorder and determined the reading to be approximately 24.43 ma which corresponds roughly to a 20% limiter setting. This is an appropriate limiter setting for a running recirc pump. Other limiter settings were as follows, "A" recirc pump: Lo-20; HI-97, "B" recirc pump: Lo-98; HI-100, "C" recirc pump: Lo-100; HI-100, "D" recirc pump: Lo-100; HI-100, "E" recirc pump: Lo-20; HI-100. "A", "D", and "E" pumps were running at the time.) Upon recirculation pump trip, the low limiter setpoint becomes the controlling input to the blind controller. After approximately 6 minutes, the controller output signal step increases offscale which might correspond to raising the low limiter setpoint up to 100. After the event, the "B" pump limiter setpoint was discovered to be set at 98. Possibly there was some consideration given to restarting the "B" pump after the trip as one would raise the limiter setpoint up to 100 to warm MG set hydraulic fluid prior to restarting the pump. This is a plausible explanation for the strip chart recorder trace. Others were explored, but this one was found to be most suitable.
02:18:14	Analysis	"C" recirculation pump tripped. "C" recirc loop flow starts to coastdown. "C" motor generator set motor winding temperatures start decreasing. "C" pump motor winding temperatures start decreasing from approximately 137° F to about 90° F over a period of hours. (Control room recorder)
		Core delta P goes to zero. If all pumps are tripped at this time, fuel zone level turned on, but does not come onscale until later.
02:18:32	Analysis	RBCCW common heat exchanger outlet pressure increases approximately 7 psig in pressure step increments corresponding to the time V-5-166, V-5-709 were shut.

Time	Source	Event
02:18:34	SAR	Reactor level high alarm 170" yarway. Level increase due to recirc pump trips and reduction in letdown flow.
02:18:36	SAR	Reactor level high alarm
02:18:38	SAR	"A" recirc pump CCW low flow due to isolation of V-5-166 and V-5-709
02:18:39	SAR	"D" recirc pump CCW low flow
02:18:50	SAR	"B" recirc pump CCW low flow
02:18:51	SAR	"C" recirc pump CCW low flow
02:19:04	Analysis	"D" recirculation discharge reaches full stroke and is open. (Valve stroke time: 96 seconds) This estimated time is based on the valve being opened at 02:17:28. Note that if "C" had been open at this point the "Less than 2 LPS" alarm would have cleared.
02:19:12	Analysis	"B" recirculation discharge valve closes. (Valve stroke time: 107 seconds; electrical backseating time allowance: 5-6 seconds.) This time is based on the valve starting to close at 02:17:20.
02:19:13	Analysis	Probable "C" recirculation discharge valve closure. (Valve stroke time: 106 seconds; electrical backseating allowance: 5-6 seconds.) The operator probably now reopened the valve as the "Less than 2 LPS" annunciator would be still illuminated and "C" loop is more preferable to reopen as "B" loop has RWCU on the loop. "C" has no other systems tied into loop piping.
02:19:17	SAR & Analysis	Less than 2 loops isolated normal alarm. At least 2 recirculation loops are open. "A" recirculation discharge isolation valve reaches full open position and clears the "Less than 2 LPS" annunciator. Stroke times are essentially based on the amount of time it takes the valve to travel from the 95% open limit switch to the full close position. The licensee estimated approximately 0.3 seconds travel time from the full open position to the 95% open limit switch under the no recirculation flow and unbackseated condition.

Time	Source	Event
02:20	Licensee Operator Interviews	Cleanup letdown flow stopped. Commenced raising Rx water level to 185 inches for natural circulation
02:20:24	Analysis	"C" recirc suction temp decreases from 144 to 142° F. This occurs as shutdown cooling flow backflows in the loops. "C" recirc discharge valve is approximately 66% open.
02:20:34	Analysis	Fuel zone water level comes onscale. Fuel zone water level is turned on with no recirculation pumps running but only indicates core region water level below 180" TAF. Considering plant conditions with level transients from turning off pumps and opening valves, it was not considered unlikely that this instrument would take nearly two minutes to indicate onscale as level drops accordingly.
02:20:59	Analysis	"C" recirculation discharge isolation valve reaches full open position.
02:21:30	Analysis	"C" recirculation loop temperature decreases from 142 to 137° F. This occurs as shutdown cooling flow backflows in the loops and possible cold feedwater flow distribution. Condensate flow is increased from ≈ 71.5 x 10^4 lb/hr to 80.35 x 10^4 lb/hr. This likely resulted from operator action to open a bypass valve around a main feed regulating valve to raise reactor vessel level to 185" to facilitate shutdown cooling flow circulation.
02:21:32	Analysis	"A" recirculation loop suction temperature decreases from 143 to 139° F.
02:22:16	Analysis	"D" recirculation loop suction temperature decreases from 142° F to 138° F.
02:24:24	SAR	
02:24:57	PSMS Computer Graph	Reactor water level Hi/Lo Alarm (170"; GEMAC)
02:30	Interviews	V-5-167 Backseated; RBCCW leak stops.
02:44	Analysis	"B" recirc pump discharge valve opened.

Table 1

Time	Source	Event
02:54:49	SAR	V-5-166 opened. RBCCW isolation alarm normal.
02:58:19	SAR	RCP "C" CCW Lo flow alarm normal. V-5-709 opened restoring RBCCW to RCP's
02:58:29	SAR	RCP "B" CCW Lo flow normal
02:58:31	SAR	RCP "D" CCW Lo flow normal
02:58:39	SAR	RCP "A" CCW Lo flow normal.
03:08:41	SAR	"D" recirc pump started.
03:15:03	SAR	"A" recirc pump started; normal shutdown conditions restored.

APPENDIX A

PROCEDURE	TITLE	REV.	EFFECTIVE DATE
700.1.030	Generic Repack Procedure for the use of Chesterton style 5300 style one (1) packing	3	8/21/87
Standing Order 33	Backseating/Unbackseating of	4	3/23/87
700.2.012	Electrically Backseating Station Valves	4	5/06/85
A100-SMM-3917.06	Manually backseating station valves	0	11/13/86
107	Procedure Control	32	8/14/87
108	Equipment Control	38	6/08/87
301	Nuclear Steam Supply System	39	3/29/87
2000-ABN-3200.19	RBCCW Failure Response	4	11/22/86

Bibliography for Section 5.2

1. Engineering Request ET. No. 391 issued 7/31/79

2. Letter from Finfrock to Eisenhut dated 6/23/80
   Letter from Finfrock to Director NRR dated 4/3/83
   Design Criteria 391-80-3, 9/12/80
   Engineering Evaluation 391-80-1, 7/29/80
   Installation Spec 391-80-4, 3/20/81
   Modification Proposal 391-80-2, 3/18/81
   Installation Spec 391-80-5, 12/10/81 (BA 402207)

3. Letter from R. F. Wilson to J. Zwolinski dated 9/19/85 (RFW-0614)

4. Letter from R. F. Wilson to J. Zwolinski dated 1/30/86 (RFW-0770)

5. Letter from J. Donohew to P. Fiedler dated 7/15/86

APPENDIX B

SEP 15 1987

MEMORANDUM FOR: William F. Kane, Director
Division of Reactor Projects

FROM: William T. Russell
Regional Administrator

SUBJECT: AUGMENTED INSPECTION TEAM - SAFETY LIMIT VIOLATION AT OYSTER CREEK

You are directed to perform a prompt inspection of the causes, safety implications, and associated operator actions during a radioactive coolant spill and subsequent actions which resulted in a Safety Limit Violation at Oyster Creek on September 11, 1987. The inspection shall be in accordance with NRC Manual Chapter 0513, Part III, and additional instructions in this memorandum.

DRP is assigned to conduct this inspection and Dr. L. H. Bettenhausen is designated as the Team Leader. The team will also include the provision for participation by the Office of Investigation and NRR.

OBJECTIVE

The general objectives of the AIT are to:

a. Conduct a timely, thorough, and systematic inspection related to the circumstances surrounding the Safety Limit Violation.

b. Assess the safety significance of the events and communicate to Regional and Headquarters management the facts and safety concerns related to the problems identified.

c. Collect, analyze, and document all relevant data and factual information to determine the causes, conditions and circumstances pertaining to the events.

d. Evaluate the adequacy of the licensee's internal review of the event.

SCOPE OF THE INSPECTION

The AIT response should identify and document the relevant facts and determine the probable causes and should be limited to the issues directly related to the Safety Limit Violation event and operator responses.

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SEP 15 1987

Specifically, the AIT should:

a. Develop a chronology of the event.

b. Determine the scope and quality of licensee's internal review of the event.

c. Develop a history of the Safety Limit and why it exists.

d. Review the maintenance activities leading to the event.

e. Determine operator response.

SCHEDULE

The AIT shall be dispatched to the Oyster Creek Nuclear Generating Station not later than September 12, 1987, and shall remain there as long as necessary to accomplish the objectives of this inspection. It is expected that this will take no longer than five working days.

A written report on this inspection will be provided to me by September 30, 1987. It may be necessary to report on operator integrity issues at a later date as they are explored and identified.

TEAM COMPOSITION

The assigned team members are as follows:

L. Bettenhausen, DRP, RI
C. Cowgill, DRP, RI
W. Bateman, SRI - Oyster Creek
J. Wechselberger, RI - Oyster Creek
D. Alsopp, RI - Hope Creek
A. Dromerick, NRR

W T Russell

William T. Russell
Regional Administrator

SEP 15 1987

cc:
J. M. Allan, DRA
W. V. Johnston, DRS
T. T. Martin, DRSS

bcc:
Team Members
S. Collins
R. Blough
W. Baunack

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

APPENDIX C

11 SEP 1987

Docket No. 50-219
Confirmatory Action Letter No. 87-12
EA No. 87-185

GPU Nuclear Corporation
ATTN: Mr. P. R. Clark
President
100 Interpace Parkway
Parsippany, New Jersey 07054

Gentlemen:

Subject: Confirmatory Action Letter 87-12 - Violation of Technical Specification Safety Limit and Subsequent Operator Actions

Pursuant to the telephone conversation on September 11, 1987 between Mr. Samuel Collins of my staff and Mr. Ed Kintner of your staff, it is our understanding that you will:

1. Maintain the plant in a shut down condition per the requirements of 10 CFR 50.36 which require Commission approval for restart.

2. Establish and maintain as found conditions for relevant hardware and process systems which pertain to analysis of the event. These conditions will be maintained until released by the NRC.

3. Conduct an independent review of the circumstances surrounding the events and forward a report to the Commission per the requirements of Technical Specification Section 6.7.

It is our understanding that upon completion of formulation of your methodology to assess the violation of Technical Specification Safety Limit events and subsequent operator actions, these plans will be discussed with the NRC prior to implementation.

If your understanding of the actions to be taken differs from the above description, please contact this office within 24 hours of receipt of this letter and before restarting the plant from its current outage.

Sincerely,

James M. Allan

William T. Russell
Regional Administrator

8710020114 2pp

APPENDIX D

SEP 14 1987

Docket No. 50-219 (CAL)
Confirmatory Action Letter 87-12
EA No. 87-185

MEMO:

To: Mr. P. B. Fiedler
Vice President and Director,
Oyster Creek

Subject: Modification of CAL 87-12

I have been authorized to release all plant equipment which was to be maintained as found by Condition 2. of the subject letter. These as found conditions are no longer needed to analyze the event.

Lee H. Bettenhausen
Leader, Augmented
Inspection Team

8711200132

APPENDIX E

Time Correction

PSMS computer was approximately 5 minutes and 49 seconds different from new plant computer time, which is considered accurate as it is updated via satellite. The licensee conducted a time check at approximately 1000 on the morning of 9/11/87 of the PSMS computer, new plant computer and the sequence of alarms recorder (SAR) to establish their accuracy using the new plant computer as a reference. Previously, the PSMS computer was restarted and as a consequence its time updated. Therefore, when time check was conducted, the PSMS computer time was approximately 30 seconds ahead of the new plant computer, but at the time of interest for the event, a 5 minute and 49 second time discrepancy existed. The SAR time was 7 seconds ahead of the new plant computer time. There was no reason to believe that either the SAR or new plant computer time had been changed.

APPENDIX F

EXIT ATTENDANCE OYSTER CREEK 9/17/87

Name	Title	Company Affiliation
A. W. DROMERICK	Lic Proj. MANAGER, Oyster CREEK	NRR-NRC
J J Rogers	Licensing Engineer	GPUN
J.D. KOWALSKI	OC Licensing Mgr	GPUN
R.P. COE	Director Training & Education	GPUN
J.L. SULLIVAN JR.	PLANT OPERATIONS DIRECTOR	GPUN-OC
MJ Slobodien	Radiological Controls Director	GPUN-OC
MB Roche	VP & Director Quality & Radiological Controls	GPUN
H.J. LAMM, JR.	Manager, Plant Training - OC	GPUN
R. FITTS	QA AUDITOR	GPUN
D. MACFARLANE	OC SITE AUDIT MGR.	GPUN
R. Feuti	Mgr - QA Mod/Ops	GPUN
E. Fuller	Operations QA Mgr	GPUN
Michael Dagort	Mgr. BWR Licensing	GPUN
T.R. BAYTER	ATTORNEY	SP&T
D Zannoni	ENGINEER	NJ - DEP.
L Hampersky	Nuclear Engineer	N.J./DEP-BNE
J.A. CAMIRE	PLANT ANALYSIS MGR	GPUN
W.A. GARVEY	MGR SPECIAL PROJECTS OC	GPUN
T.M. OSBORN	NSCC Staff	NUS Corp
A.T. MOROWEY	NSCC Staff	NUS
R.D. DAVIDSON	GPUN - OPERTNG MGR - OC.	GPUN
D.J. RANTT	MGR. - PLANT ENGR.	GPUN
K. Eckeissen	Mgr. Nuclear Safety	GPUN (part time)

EXIT ATTENDANCE OYSTER CREEK 9/17/87

Name	Title	Company Affil.
S Polon	MGR. - PUBLIC Information	GPUN
P. L. Long	Director Planning & Nuclear Safety	GPUN
P.B. FIEDLER	DIR OC	GPUNC
J.J. BARTON	DEPUTY DIRECTOR OCNGS	GPUN
MB Roche	Director Qual. + Rad Con	GPUN
D.K. CRONEBERGER	DIRECTOR - ENGINEERING PROJECTS	GPUN
JR Thorpe	Director - Licensing & Reg. Affairs	GPUN
C R Tracy	Director Engr. Assur	GPUN
D.K. ALLSOPP	RI, HOPE CREEK	NRC
J.F. WECHSELBERGER	RESIDENT, O.C.	NRC
W.H. Bateman	SRI, O.C.	NRC
L. Bettenhausen	Team Leader, AIT	NRC

cc:
P.B. Fiedler, Vice President and Director
M. Laggart, BWR Licensing Manager
Licensing Manager, Oyster Creek
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New Jersey