CP&L

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February 25, 1991

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BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
LICENSEE EVENT REPORT 2-91-001

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

R. L. Warden for

J. W. Spencer, General Manager
Brunswick Nuclear Project

TH/

Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
BSEP NRC Resident Office



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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Brunswick Steam Electric Plant Unit 2 | 05000324 | 01 OF 19 |

TITLE (4) Unit 2 Turbine Trip/Reactor SCRAM While Calibrating a Feedwater Computer Point

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQ. NO. | REV. NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 01 | 25 | 91 | 91 | - 001 - | 00 | 02 | 25 | 91 | | |

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

| OPERATING MODE (9) | 1 | | | |
|---|---|---|---|---|
| POWER LEVEL (10) | 100 | | | |
| | 20.402(b) | 20.405(c) | X 50.73(a)(2)(iv) | 73.71(b) |
| | 20.405(a)(1)(i) | 50.36(c)(1) | 50.73(a)(2)(v) | 73.71(c) |
| | 20.405(a)(1)(ii) | 50.36(c)(2) | 50.73(a)(2)(vii) | OTHER (Specify in Abstract and Text) |
| | 20.405(a)(1)(iii) | 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| | 20.405(a)(1)(iv) | 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | |
| | 20.405(a)(1)(v) | 50.73(a)(2)(iii) | 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER |
|---|---|
| TONY HARRIS, REGULATORY COMPLIANCE SPECIALIST | (919) 457-2038 |

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

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| X YES (If yes, complete EXPECTED SUBMISSION DATE) | NO | | 04 | 01 | 91 |

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


On January 25, 1991, Unit 2 reactor scrammed from 100% power. The scram was due to a turbine trip on high reactor water level which resulted from the Feedwater Level Control System responding to a sensed loss of feed flow during performance of a Process Computer point calibration on the Feedwater flow logic system. Instrumentation and Control (I&C) technicians performing the calibration failed to recognize a procedure prerequisite step which stated that the unit must be in cold shutdown or refuel to perform the procedure.

The event was due to the failure of the work control process to prevent this activity from being performed, caused by inadequate reviews in the scheduling and implementation phases of the process and the failure of the technicians to ensure the prerequisites were met. In addition, the procedure Summary Sheet incorrectly stated that no special plant conditions were required for the performance of the procedure. Corrective actions included stoppage of the computer point calibrations, suspension of use of the procedure Summary Sheets, communication of the event with plant personnel, development of a Recovery Action Plan, and initiation of a working level Task Force to investigate this and similar recent personnel errors. Future corrective actions will be identified in a supplement to this LER pending completion of management review of the HPES and Task Force evaluations.

The safety systems functioned as designed. Equipment response concerns were identified, but none posed a significant safety concern.

EVENT

Unit 2 Turbine trip/Reactor SCRAM during calibration of Feedwater Process Computer point B022.

INITIAL CONDITIONS

Unit 2 was operating at 100% power. Reactor pressure was at 1010 psig, and reactor vessel level was at 186 inches. Reactor Feedwater Level Control was in automatic, three element control. The following systems were operable in standby readiness: High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Automatic Depressurization System (ADS), Reactor Protection System (RPS), Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI), Core Spray (CS), Standby Gas Treatment (SBGT), Standby Liquid Control (SLC), and Emergency Diesel Generator (EDG). A Process Computer point instrumentation calibration was in progress on computer point A1713 (B022), Feedwater Flow Loop A, in accordance with Process Instrument Calibration (PIC) OPIC-CPU001, Attachment 13 Data Sheet. Instrumentation and Control (I&C) technicians, in accordance with Step 7.1.1 of OPIC-CPU001, were preparing to lift wire number C32-A-18 from terminal DD-84.

EVENT DESCRIPTION

On January 25, 1991, at 0810, the technicians lifted wire C32-A-18 from terminal DD-84 in the Feedwater logic loop. A detailed sequence of events for the resulting transient is provided in Attachment A. The lifting of wire C32-A-18 caused a loss of the "A" Feedwater flow signal into the Feedwater Level Control System (FWLCS). As a result, the FWLCS increased the speed of both Reactor Feed Pumps (RFP). This caused an increased flow in the feedwater path, which increased Reactor Water Level to the High Level Turbine Trip setpoint. The Main Turbine and both turbine driven RFPs automatically tripped. As a result of the Turbine Trip, a Turbine Stop Valve closure occurred, initiating an automatic Reactor Scram.

Following the Reactor Scram, vessel pressure increased to a transient maximum of 1032 psig. Reactor Water level decreased to a transient minimum value of between 116.5" and 121", approaching the Low Level 2 (LL2) instrument trip setpoints. As a result, both the HPCI and RCIC systems initiated, both Recirculation pumps tripped, and Groups 2, 6, and 8 isolation signals were generated. A partial Group 3 isolation signal was also received. RCIC began injecting into the reactor vessel, and HPCI entered into a minimum flow pathway, not injecting into the reactor vessel. Level was being restored to above the LL2 setpoint, and the Control Operator manually initiated HPCI to assist in restoring vessel to normal level. HPCI and RCIC were secured once vessel level was returned to normal range. The isolations were reset at 0822, and all rods were confirmed to be fully in. Normal recovery procedures were then followed. The Site Incident Investigation Team (SIIT) was convened to begin investigation of the event.

EVENT INVESTIGATION

The event investigation initially focused on determining the origin of the feedwater flow mismatch. Review of ongoing work activities determined that the calibration being performed on the Process Computer Feedwater flow point was a possible origin of the Feedwater flow mismatch. Interviews with involved personnel and reviews of the controlling procedure determined that a Prerequisite step in the procedure (Step 3.2.1) requires that "the unit will have to be in cold shutdown or refuel condition to perform this test, due to the interlocks and controls of this loop (feedwater and recirculation)." Additionally, the Precautions and Limitations section of the procedure identifies the involved instrument as part of the single point failure analysis.

The I&C technicians performing the procedure lifted a wire in the feedwater control system as specified by Step 7.1.1 of the procedure. The wire removal resulted in the loss of part of the feedwater flow signal to the reactor level control system and created a false steam flow/feedwater flow mismatch. The level control system responded as designed by increasing feedwater flow to the reactor to compensate for the sensed reduction in feedwater flow. This action increased the reactor feed pump speeds and actual feedwater flow. The increased flow increased reactor water level until the high reactor water level trip setpoint was exceeded. The reactor high water level trip initiated trips for the turbine driven reactor feedpumps and the main turbine. The main turbine trip initiated the reactor SCRAM.

The immediate investigation performed by the SIIT was initiated to determine the root cause of the SCRAM, and to reconcile potential problems associated with the SCRAM and recovery evolutions. Additional investigations, including a Plant Incident Report and Human Performance Enhancement System (HPES) evaluation, are being completed. The primary factors in the root cause analysis are presented below.

ROOT CAUSE ANALYSIS

The January 25, 1991 SCRAM was the result of the performance of a preventive maintenance calibration procedure (OPIC-CPU001) during operation which had prerequisites mandating that the procedure be performed during cold shutdown or refuel conditions. The procedure involves the calibration of a signal to the process computer from the feedwater control system. Since performance of this procedure affects the operation of the feedwater loop flow input to the reactor level control system, the procedure should only be performed while the unit is shutdown or in the refuel condition.

The work control process for a preventive maintenance activity at the Brunswick plant involves three phases: procedure preparation and preventive maintenance (PM) route (a prearranged sequence for performing the PM) development, work scheduling, and work implementation. For this event, each phase of the work control process had primary barriers that failed.

The procedure prepared for this route was inaccurate in that the Attachment Summary Sheet incorrectly stated that the procedure could be performed under any plant condition. The reviews of this procedure revision did not detect the inconsistency between the Summary Sheet and the Prerequisites in the body of the procedure.

The work scheduling phase did not keep or remove this item from the work schedule. This phase involves multiple reviews of potential work items to ensure that a particular item can be performed at the specified time given the anticipated plant conditions. The inadequacy of the reviews which support this phase led to the route being scheduled for work.

The work implementation phase of the work control process failed to prevent the calibration from being performed. The reviews associated with this portion of the process did not detect the prerequisite plant condition for performing this calibration. In addition, the involved technicians did not follow the calibration procedure when they failed to ensure the procedure prerequisites were satisfied prior to lifting the wire.

The focus of an operating unit work scheduling phase at the Brunswick plant is the Site Work Force Control Group (SWFCG). Work items are scheduled through the SWFCG, which includes representatives from each site work organization. SWFCG, however, relies on a multitude of reviews to ensure that work presented to the group is acceptable for a given plant condition. For a typical maintenance work item, such as this process computer point calibration, the scheduling process is as follows:

1. The planning process for preventive maintenance activities begins with an automatic computer function which prompts a maintenance planner/analyst from an interval based computer display to generate a route sheet WR/JO. The route sheet lists affected components and procedure numbers necessary to complete the route. The planner does not review procedures or the effects that component manipulation will have on the plant. The designated route sheet is then given to the responsible maintenance foreman for initial screening, procedure review, and scheduling.

2. The responsible maintenance foreman performs the initial assessment of the item to determine plant conditions required for the item to be performed, and at what time these conditions will exist. The foreman reviews the work item, including the applicable system work schedule, prerequisites and precautions involved for performing the job, and plant conditions required for an item to be worked. Once the foreman determines that an item can be worked with current plant conditions, he routes a package to the SWFCG describing the item, the system(s) affected, and the preferred work time if no specific system constraints exist.

3. The SWFCG scheduling coordinator develops a system-sorted list of potential work items to be reviewed by the SWFCG at a weekly input meeting. Each system sort is uniquely reviewed to determine the appropriate time for scheduling the involved work. Questions are raised by participating groups if the input sheets do not provide sufficient information to determine potential plant affects from a proposed work item. Once an item is approved by the SWFCG for work, the item is placed on a work schedule for a given week.

4. Once scheduled, items on the weekly schedule are reviewed by the Control Room Operations organization prior to the scheduled performance to ensure that the item remains acceptable for the current plant conditions. Once approved for work on a given day, the work package is given back to the responsible foreman to implement.

Implementation of a preventive maintenance activity begins with the responsible maintenance foreman. The responsible foreman for the job activity plans the work to be done on a given day. Part of the responsible foreman's implementation function is to conduct a pre-job briefing with the technicians performing the job. The discussions with the technicians include the precautions and limitations associated with a given task. The technicians are then given the package to perform.

The work package given to the technician includes the WR/JO developed for the job. The WR/JO is taken to the Control Room, if necessary, to obtain Shift Foreman approval for beginning the job. Once the Shift Foreman has approved the job start by reviewing the package, the job is begun by the technicians. The technicians performing the job complete the job in accordance with the process defined by the WR/JO.

The work control process at the Brunswick plant during plant operation is a multi-level, multi-organization review and implementation process, involving several individuals. The process is dependent upon the completion of these reviews and activities in accordance with established standards and procedures. The standards and procedures for these reviews and activities, if properly followed, would have ensured that this procedure would not have been performed. However, the incorrect reliance by some of the involved individual reviewers allowed a single factor, the incorrect Summary Sheet, to defeat the barriers established by the system. The January 25, 1991 SCRAM was, therefore, the result of the failure of the reviews in this process to identify the required conditions for performing this job. A summary of the barriers in the work control process and the failures within each barrier follows.

One of the causal factors contributing to the inadequate reviews was the universal use of an incorrect procedure Summary Sheet. The body of the procedure being used (OPIC-CPU001) has a Prerequisite Section which correctly specifies that the procedure should only be performed while the plant is in cold shutdown or refuel condition. The Summary Sheet, provided as an attachment to the procedure, summarizes the impact of the procedure and includes such items as a procedure description, required plant conditions for performance of the procedure, alterations to plant systems as a result of the procedure, annunciators and affected indications, and possible Technical Specification Limited Conditions for Operation (LCOs) which may result from the performance of the procedure. The Summary Sheet is intended to provide an impact summary for the Shift Foreman to use in assessing whether the procedure can be performed under the conditions which exist at the time the procedure is planned to be worked by the Maintenance organization. The Summary Sheet is not intended to be substituted for the Prerequisites and Precautions stated in the procedure, and states so in the heading of the page. The Summary Sheet for this procedure incorrectly indicated there were no special plant conditions required for performance of the procedure, and this greatly influenced the mind set of the involved individuals both performing the procedure and reviewing the procedure for possible plant impact.

The initial barrier in the scheduling phase that was defeated was the I&C foreman review of the work item. The I&C foreman initially submitted six similar calibration routes to the SWFCG for work. The calibration of computer point B022 was included in this package. The foreman had not appropriately reviewed the precautions and limitations for each package prior to submitting the SWFCG work request as required by Maintenance procedures.

Once the calibration routes were input into the SWFCG system index, the items were reviewed at the weekly SWFCG "upcoming work" input meeting. Questions were raised by the cognizant Operations representative concerning these calibrations, and the Maintenance representative was requested to further investigate and provide input to the SWFCG on plant affects from these calibrations. The Maintenance representative reviewed the procedure Summary Sheets for the work and determined that two of the routes required the plant to be in either shutdown or refuel condition to perform. He then returned the packages to the responsible I&C foreman to reevaluate the remaining packages for additional impact.

The I&C foreman's second review of the work packages consisted of a review of the procedure Summary Sheets for the involved procedures. This was not an appropriate review of the involved work as dictated by Maintenance practices. Had the foreman reviewed the Prerequisites and Precautions section of the procedure during either his first or second review, as defined by existing Maintenance standards, he would have noted that the Prerequisites of the procedure for the calibration of computer point B022 required the plant to be in either a shutdown or refuel condition.

The I&C foreman, following completion of his second review, returned the work package to the Maintenance SWFCG representative. The package included a copy of the Summary Sheet for the calibration of computer point B022. SWFCG review appropriately credited the work done by the I&C foreman, and thus relied on an inaccurate Summary Sheet in determining that the work was safe to be performed under the existing plant conditions.

Once the calibration was placed on the SWFCG work schedule for the week of January 19 through January 25, 1991, the package was given to the Operations staff for a final review. The Senior Reactor Operator (SRO) reviewing the work for the upcoming day noted the required conditions as defined by the SWFCG package and the Procedure Summary Sheet. The SRO identified a potential concern about the affects of the evolution on the Process Computer Periodic Core Performance Log (P1), which monitors core performance parameters. A note was put on the package for the Technicians working the job to contact the dayshift Shift Foreman to identify the affects on P1 from this activity.

The implementation phase of the work control process began on Friday morning, January 25, 1991. The I&C technicians received a pre-job briefing from the responsible I&C foreman for the calibration of the computer point. The technicians noted that the responsible I&C Foreman instructed them that this was a job with no plant impact. The I&C foreman pre-job briefing did not include a review of the prerequisites and precautions section of the procedure, as required by Maintenance practice. The technicians proceeded to the Control Room to discuss the P1 note with the Shift Foreman.

The Shift Foreman had discussed the affects of similar computer points with the Nuclear Engineering group earlier that week. The affects of these jobs were determined to be of no concern relative to the P1 Process Computer log. The Shift Foreman then reviewed the Summary Sheet, which stated that there were no plant required conditions, and plant systems being altered were limited to making computer points on the Process Computer and Emergency Response Facility Information System (ERFIS) inoperable. The Shift Foreman signed the package and instructed the individuals to obtain the concurrence of the Senior Shift Foreman prior to starting work. The Senior Shift Foreman was also misled by the Summary Sheet into believing that the job would not have any affect on the operating unit.

The final barrier in the implementation phase was the I&C technicians performing the calibration. The two I&C technicians assigned to perform the calibration focused on the procedure Summary Sheet for assessing the plant impact of performing the procedure. The mindsets of the technicians involved with the event were the result of the work on computer points earlier in the week, the incorrect Summary Sheet statement, and the pre-job briefing held with the I&C foreman just prior to starting the job. As a result, the technicians did not ensure the prerequisites of the procedure were satisfied prior to initiating the calibration. The prerequisite step which requires the plant to be in cold shutdown or refuel condition prior to performing the procedure immediately follows a sign-off for shift foreman approval to begin the procedure. This step was overlooked by the technicians. As the final barrier, the failure of the technicians to follow the procedure prerequisite step directly resulted in the SCRAM.

In summary, the work control process of the operating plant did not eliminate a work item that should have been performed only with the plant in a shutdown or refuel condition. The process failure was a result of the inadequate reviews relied upon in the scheduling process to ensure work items are performed only during desired modes of operation, and a failure of the involved technicians and foreman to ensure prerequisites of the procedure were satisfied. The inadequate reviews were due to a combination of personnel failures to adhere to established procedures and directives, and an incorrect procedure which led the reviewers to believe that this particular calibration could be performed during unit operation.

ABNORMAL TRANSIENT OCCURRENCES

This section provides a summary of any system or component failures experienced, as well as explanations of unusual occurrences that occurred during the plant transient.

The following items were reported in the initial red phone report as potential concerns that needed further investigation:

1. Standby Gas Treatment (SBGT) System not starting and the Reactor Building ventilation dampers not closing.

2. Reactor Water Cleanup (RWCU) Group 3 isolations not occurring.

3. HPCI F006 valve not automatically opening and injecting.

4. Possible excessive closure time of the G16-F020 Drywell Equipment Drain Outboard Isolation Valve.

Resolution of these items is discussed below. Additional concerns found during the event are also discussed.

<u>SBGT SYSTEM ACTUATION, GROUP 3 AND RB VENT DAMPER ISOLATIONS</u>

SBGT system actuation, Reactor Building (RB) Ventilation damper isolation, and a Group 3 Primary Containment Isolation System (PCIS) valve isolation are initiated by the same instrumentation group for Reactor Vessel Water Level Low level. The involved instrumentation (four instruments: 2-B21-LT-N024A-1, A-2, B-1, and B-2) provide trips at Reactor water level 118", decreasing. The logic is such that an "A" device must trip in conjunction with a "B" device to initiate the isolation or actuation. The as-left setpoints of the four trip instruments in this logic system range from 117.6" to 117.8". A review of the data from the January 25, 1991 SCRAM for the wide range instrumentation determined that reactor water level dropped to between 116.5" and 121". Based on the fact that the level did not positively decrease to less than the isolation/actuation setpoints during this event, and that simultaneous actuation of both the "A" and "B" channels would have to occur for the isolations/actuations to be completed, it is not considered unusual for the isolations and actuations to not take place as a result of the transient conditions resulting from this event. It is probable that more than one of the individual instruments did not concurrently trip to initiate the Group 3 and Reactor Building damper isolations, as well as SBGT system actuation; therefore, based on the predicted instrument responses and the lowest vessel level seen during this transient, these items are not considered unusual responses.

<u>HPCI/RCIC OPERATION</u>

A review of the Emergency Response Facility Information System (ERFIS) data associated with the operation of the HPCI and RCIC systems for this event has determined that the systems performed as expected, with no concerns noted. The HPCI system automatically started, but did not inject into the vessel. This is an expected response due to the short duration (< 5 seconds) of the Low Level 2 (118" trip, 122" reset) initiation signal. HPCI operated in the minimum flow mode until the F006 HPCI injection valve was manually opened for reactor level recovery, approximately one minute following the initiation signal. HPCI injected for approximately 2 minutes. Following level recovery, HPCI was transferred to the Pressure Control mode. HPCI was manually removed from service after a total operation time of 6.5 minutes.

The RCIC system automatically started and injected into the vessel, as designed, in response to the Low Level 2 actuation signal. RCIC injected at 400 gpm until level was restored (approximately 4.5 minutes). RCIC flow was manually reduced over a period of four minutes. RCIC was manually secured and placed in standby readiness after a total operation time of 8.5 minutes.

2-G16-F020 ISOLATION TIME

The Group 2 isolation signals received during this event initiate closure of the Drywell Floor and Equipment Drain Inboard and Outboard Isolation Valves (2-G16-F003, F004, F019, and F020). Closure of the F003, F004 and F019 valves occurred within the expected closure times (approximately 3.5 seconds). The F020 Equipment Drain Outboard Isolation valve did not indicate full closed until approximately 11 seconds after receiving the closure signal. The closure time was determined to be excessive compared to the other valves and exceeded the acceptance criteria found in Periodic Test (PT)-11.3, Drywell Drains System Valve Operability Test. The closure time was within the Technical Specification operability requirement of 20 seconds.

An outstanding Work Request/Job Order (WR/JO) exists for replacing the limit switch for the 2-G16-F020 valve. Troubleshooting has determined that an intermittent problem with the switch causes a delay in the closed indication signal being received, due to a binding condition with the switch protective boot over the switch plunger. Periodic Test (PT)-11.3 was performed to ensure the valve stroke time was within Technical Specification limits and testing acceptance criteria. The testing determined that the valve stroked in approximately 3.5 seconds, by indication. The switch worked appropriately during the testing.

The excess closure time of the F020 valve seen during the transient was determined to be the result of the intermittent problem with the valve limit switch. This is not believed to be an operability concern; however, parts are on order to facilitate the necessary repair.

2-B32-F031B VALVE

Following the SCRAM, an attempt was made to restart the 2B Reactor Recirculation pump. The pump would not start as a result of Motor Generator Set 2B field breaker not opening. The 2-B32-F031B Reactor Recirculation Pump discharge valve was being opened following the failed start attempt, when the valve bound up at approximately 70% open.

The 2-B32-F031B valve is a normally open valve, but is closed in order to place a Recirculation pump in operation. The F031B valve is then opened after the pump is restarted. The valve failure occurred as the valve was being opened following the failed 2B pump restart attempt. WR/JO 91-ABRF1 was initiated to investigate this problem.

An entry was made into the Unit 2 Drywell to inspect the F031B valve. The valve showed signs of a packing leak of approximately 120-140 drips per minute, and an inspection of the valve stem indicated galling near the packing gland. WR/JO 91-ABTT1 was initiated to further investigate and repair the valve. A valve manufacturer representative (Anchor Darling) assisted in the investigation and repair of the F032 valve.

The stem of the F031B valve was found galled into the valve stuffing box area. The gall marks ran parallel with the stem travel and were located 180 degrees apart, oriented with the pipe run. The gall marks covered a distance on the stem of approximately 20" on both the upstream and downstream sides of the stem.

Based on available visual information and discussions with the valve Vendor, the BSEP Technical Support unit has determined the most probable root cause of the stem galling and subsequent valve failure to be the result of stem contact with the packing gland. The contact was most likely caused by gland misalignment and/or an introduction of foreign material into the gland/stem area. The initial galling occurred on one side of the stem, the downstream side, due to system flow and pressure. Stem galling on the upstream side of the stem was the result of a decreased radial clearance caused by the accumulation of metal shavings. Subsequent valve stroking would have increased the metal deposit in the failed area(s) of the stem to the point that the valve actuator would not deliver the necessary torque to open the valve. The valve travel thus stopped as a result of the provided over-torque protection.

The F031B valve is closed each time the Recirculation pump trips to facilitate pump restart. The valve has been cycled approximately six times since the last refueling outage. The remaining three Recirculation pump inlet and discharge valves were inspected, with no signs of galling noted. No instances of stem galling associated with the new single stage packing have been noted. The valves will be inspected during the upcoming Unit 2 outage to ensure that the root cause of the valve stem galling has been adequately assessed.

<u>RHR E11-F003A VALVE FAILURE DURING SHUTDOWN COOLING</u>

During shutdown cooling following the event, the RHR E11-F003 RHR Heat Exchanger Outlet Valve breaker tripped while adjusting the cooling flow during shutdown cooling. This is a motor operated gate valve used by Operations during shutdown cooling operations as a throttling valve to control the RHR flow to the vessel. Upon investigation, the valve motor was found shorted to ground, which resulted in the breaker trip. The motor was subsequently replaced.

Use of the F003 motor operated gate valve as a throttling valve has been previously identified as a misapplication of use. Replacement of these valves has been identified in projects G0010B and G0010C, Plant Modifications 90-035 and 90-034 for both Unit 1 and Unit 2.

2B MG SET

A field breaker in the 4160 volt breaker compartment of the 2B Motor Generator (MG) Set was found tripped following an attempted recovery restart during the SCRAM recovery, and WR/JO 91-ABPW1 was initiated to investigate this concern.

Troubleshooting was unable to determine the cause of the breaker failing to latch. Subsequent attempts to latch the breaker during troubleshooting were successful.

2A MG SET

During the Unit 2 SCRAM recovery efforts, the 2A MG set motor breaker immediately tripped open when the recirculation pump was given a start command. Troubleshooting identified a temperature switch (mercury contact type) which was closing each time the switch contact saw vibration from the motor receiving "in-rush" current, similar to that expected during a start. The switch was repaired under WR/JO 91-ABPU1.

SPURIOUS PROCESS COMPUTER OPEN INDICATION ON SRV B21-F013K

During review of the Process Computer data for the 1/25/91 transient, the printout showed a momentary lifting of SRV B21-F013K; however, no other indications were seen that this SRV lifted during the transient. The maximum reactor pressure seen for this event would not require a SRV to open. In addition, the process computer had shown false SRV F013K opening indications prior to the SCRAM. The SRV lifting indication was therefore considered to be a false indication generated in the Process Computer logic. The reason for the false indication is being investigated.

IMMEDIATE CORRECTIVE ACTIONS

Immediate corrective actions taken as a result of this event included:

1. Capturing work in progress and organization of the SIIT to identify event anomalies and root cause.

2. Stoppage of Computer point calibrations.

3. Suspended Shift Foreman reliance on the use of Summary Sheets.

4. Communication of the event with plant personnel, including a briefing on the event and the personnel errors involved with the event.

To allow for Unit restart, a Recovery Action Plan was developed by the SIIT, including the following items:

1. Troubleshoot and resolve problems with the opening of the B32-F031B Recirculation pump discharge valve (WR/JO 91 ABTT1 & EER 91-0036).

2. Resolve problem with the 2A Recirculation pump motor breaker (WR/JO 91-ABPU1 & TSM 91-074).

3. Resolve problem with the 2B Recirculation pump start sequence (WR/JO 91-ABPW1 & TSM 91-074).

4. Restore the Unit 1 Startup Auxiliary Transformer (SAT) to service.

5. Repair the Suppression Pool Temperature Monitoring System (SPTMS).

6. Resolve the RHR F003A valve problem (WR/JO 91-ABSG1).

7. Once the above items have been satisfactorily performed, obtain duty plant manager approval for startup.

These items were completed by 1/30/91, prior to restart.

ADDITIONAL CORRECTIVE ACTIONS

In addition to the corrective actions associated with the SIIT Startup Recovery Action Plan, additional long term corrective actions were considered by the SIIT as a result of this event.

1. Investigate the spurious open indication on the process computer for SRV B21-F013K.

2. Investigate the elimination of the 40% steam flow isolation, or other action to reduce the probability of spurious Group 1 isolations due to failure of the trip unit to reset. A half Group 1 isolation was received during the SCRAM recovery on this event due to the 40% steam flow isolation (a Unit 2 function only), and has previously been identified in LERs as the contributing factor to other spurious Group 1 isolations. The isolations are not considered to be a safety concern, but may create an annoyance during operator recovery efforts following a SCRAM.

3. Investigate whether the Recirculation pump discharge valves can be shut after a Recirculation pump trip to maintain Recirculation loop temperatures and facilitate a pump restart.

4. Review and revise maintenance procedures having summary sheets to ensure that the prerequisites and equipment actuations and limitations accurately reflect the procedure impact.

5. Maintenance will review and revise the periodic maintenance (PM) routes which presently have 18 month frequency requirements to "Refuel" ("RO") frequencies if determined that plant operating conditions other than "RUN" are required for the performance of the

procedure. The plant Technical Specifications equate "18 month" frequencies with "refuel" ("RO") frequencies in terms of required timetable for the performance of surveillances/testing. Some of the Periodic Maintenance (PM) routes which are "18-month" frequencies would have no impact on the plant if performed during power operation. Other testing/calibrations, such as the one in this event, have requirements that the unit be in an operating condition other than "RUN". Designating such PM routes as "RO" would add an additional barrier against the performance of maintenance activities requiring that the plant be in a shutdown condition during operation. This designation is made by the Maintenance organization during development of a route code.

6. Brunswick Site Procedure (BSP)-43 is being issued to enhance existing SWFCC guidelines. The procedure has been strengthened to require more than one method of verifying plant impact of planned work, such as review of drawings, summary sheets, procedure prerequisites, and system descriptions.

7. A Human Performance Enhancement System (HPES) evaluation is being conducted for the personnel errors involved with this event, to determine causal factors associated with the personnel errors, as well as determining if additional corrective actions may be necessary to prevent recurrence.

8. The event and event causes are to be communicated to appropriate site personnel.

9. Investigate the feasibility of adjusting the setpoints of various actuations instruments to prevent the partial actuations that are prevalent on a high power SCRAM. This is a recurring problem during this type transient.

10. Plant Management established a Task Force compris d of working level individuals directly involved in this and other events involving personnel errors, to investigate potential underlying causes recent events involving personnel error. The Task Force composition is intended to provide Plant Management with a perspective of event causal factors from persons actually involved with an event, along with recommendations to help prevent recurrence of similar personnel errors at the Brunswick plant. The Task Force was chartered to provide recommendations which would result in plant "personnel consistently performing scheduled and emergent work activities in compliance with accurate site-approved work control process requirements and expectations." A presentation of the Task Force findings to Plant Management was made on February 22, 1991.

Upon completion of the HPES evaluation, Plant Management will review the recommendations of this effort and the Task Force to determine if additional corrective actions are necessary to help prevent recurrence of similar events. A supplement to this LER will be submitted following Plant Management review and assessment of the recommendations. The supplement will be issued by April 1, 1991, and will also provide an update and schedule for completion of outstanding corrective actions identified by the SIIT.

EVENT ASSESSMENT

The Unit was operating at a maximum power level at the time of the event. The Unit response for this event was within the bounding parameters of the corresponding FSAR Chapter 15 Feedwater Controller Failure event involving an increase in feedwater flow to the Reactor vessel. The equipment concerns identified during the event did not significantly hamper operator ability to achieve and maintain the reactor in a shutdown condition. This event would not have been more severe under any other credible and reasonable conditions.

Other SCRAMs in the past two years at Brunswick that have been directly contributed to personnel errors have been reported in LERs 2-89-09 and 2-90-09. This event is not similar to the other SCRAMs in that the work control process that did not stop the work from being performed has significant existing programmatic barriers in place to prevent such incidents from occurring, and an excellent track record for preventing this type event.

As identified in the Corrective Action section, Plant Management is disturbed with the number of plant system challenges and transient occurrences that have occurred in the past year resulting from personnel errors involving experienced and skilled personnel. The Task Force of event involved personnel is believed to be an essential step in determining causal factors involved in Brunswick personnel errors, as well as determining effective corrective actions which would reduce the frequency of occurrence of this type event.

This event was initially reported to the NRC as a one hour red phone report under 10CFR50.72(b)(1)(ii), the plant being in a degraded condition, and 10CFR50.72(b)(1)(iv), an event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal, and the four hour criteria, 10CFR50.72(b)(2)(ii), an RPS actuation, including plant SCRAM. Additionally, immediate event review determined that the event was also reportable under 10CFR50.72(b)(2)(ii), ESF actuations. These calls were made within their required reporting times.

Initial event response conservatively reported the plant to be in a degraded condition relative to potential HPCI and SBGT system failures, and that the event should have resulted in ECCS injection into the vessel from a valid signal. As described in this LER, event review indicated that the HPCI and SBGT systems operated as designed during this event, an expected result of partial instrument logic actuation and instrument permissive time constraints. Therefore the plant was not in a degraded condition at any time during this event, nor should HPCI have injected into the vessel, as designed time permissives for injection were not satisfied (approximately 13 seconds). This LER is not reporting this event relative to the plant being in a degraded condition or required ECCS actuation. The situation created by the partial actuations seen during this event and similar high power SCRAM transients that led to the initial conservative call of degradation and ECCS injection is being addressed by Corrective Action item 8.

## EIIS CODES

| SYSTEM/COMPONENT | CODE |
|---|---|
| Automatic Depressurization System | * |
| Core Spray | BM |
| Feedwater Level Control System | JK |
| High Pressure Coolant Injection System | BJ |
| Primary Containment Isolation System | JM |
| Reactor Core Isolation Cooling System | BN |
| Residual Heat Removal/Low Pressure Coolant Injection | BO |
| Standby Gas Treatment System | BH |
| Emergency Diesel Generator | EK |
| Reactor Protection System | JE |
| Reactor Water Cleanup System | CE |
| Standby Liquid Control | BR |
| Turbine Stop Valve | TA/ISV |
| Reactor Feed Pump | SJ/P |
| Startup Level Control Valve | SD/LCV |
| Process Computer | IO/CPU |
| Reactor Recirculation Pump | RR/P |
| Emergency Response Facility Information System | IQ |
| Reactor Building Ventilation Isolation Dampers | VA/ISO |
| Drywell Equipment Drain Outboard Isolation Valve | JM/ISO |
| Reactor Recirculation Pump Discharge Valve | RR/ISO |
| RHR Heat Exchanger Outlet Valve | BO/ISO |
| Reactor Recirculation Motor-Generator Sets | RR/MG |
| Safety Relief Valve | */RV |

* No EIIS System Identifier Found

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## ATTACHMENT 1

## SEQUENCE OF EVENTS--JANUARY 25, 1991 SCRAM

The following is a SEQUENCE OF EVENTS for the January 25, 1991 SCRAM. The times shown below are referenced to the Process Computer clock. Event times taken from the Emergency Response Facility Information System (ERFIS) printout have been modified by subtracting 34 seconds.

08:10:52 I&C technicians lift the wire on terminal board DD-84, which causes a loss of the "A" Feedwater flow signal into the Feedwater Level Control System (FWLCS). FWLCS increases the speed of both RFPs.

08:10:54 Computer alarm received on Condensate Filter Demineralizer high differential pressure, due to increased flow in the Condensate/Feedwater path. No annunciator was received, as the card for that window was pulled, taking the annunciator out of service.

08:11:00 Alarm received on High Reactor Water Level.

08:11:04 APRM upscale alarm received. Flux increase is due to the feedwater flow increase and cooler moderator temperature. Peak APRM reading is 106%. Condensate Booster Pump "C" automatically started.

08:11:09 Reactor Water Level reaches the high level Turbine Trip setpoint. The main Turbine and both Feedwater turbines are tripped automatically. Peak Level reached during the event is 208".

Turbine Stop Valve closure generates a Reactor SCRAM signal and rod insertion begins.

08:11:10 All four Turbine Control Valve Fast Closure (TCVCF) subchannels detect low pressure and also receive a load reject SCRAM signal (a Unit 2 function only).

Reactor high pressure SCRAM signal on channel "A2" received momentarily (approximately .5 seconds). Peak pressure observed is 1032 psig. Examination of calibration records reveals that reactor pressure trip unit B21-N023C-1 for subchannel A2 is set slightly lower than the other three subchannels, within the range of observed pressure. This response is therefore acceptable and expected.

Generator Output Breakers open.

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# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION



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ATTACHMENT 1 (CONT.)

| | |
|---|---|
| 08:11:12 | Neutron Monitoring System SCRAM signals generated on channels A2 and B1. This trip signal is the result of a combination logic of APRM downscale and IRM "Upscale/Inop". Unit 2 has been maintaining IRM F (A2) and IRM G (B1) "Inop." due to equipment problems; therefore, these neutron monitoring trips are anticipated responses when the associated APRM drops below the downscale alarm setpoint. |
| | Turbine bypass valves are fully open. |
| 08:11:17 | Loss of Feedwater turbines causes a decrease in Reactor water inventory. All four subchannels detect low reactor water level and generate scram signals. |
| | Low reactor water level generates the group 2, 6 and 8 isolation signals. The group 8 valves do not isolate, as they have been isolated prior to the SCRAM. |
| 08:11:20-21 | Valves 2-G16-F003, F004 and F019 close due to the isolation signal. |
| 08:11:22 | Following procedure, the Control Operator inserts a manual SCRAM in both channels and transfers the reactor mode switch from "RUN" to "SHUTDOWN". |
| | A half group 1 isolation signal is received in subchannel "A2". The signal is a 40% steam flow signal enabled by removing the mode switch from "RUN". Trip unit B21-N008C-2 is observed to be in the alarm state. |
| 08:11:39-43 | All four subchannels of the SCRAM Discharge Volume (SDV) level detect the HI-HI level. |
| 08:11:40 | Reactor water level reaches the Low Level 2 setpoint, and Alternate Rod Injection is initiated. |
| 08:11:46 | SDV Hi water level Rod Block is generated. |
| | At approximately this time, HPCI and RCIC receive automatic initiation commands, and the Anticipated Transient Without SCRAM (ATWS) circuitry trips both Reactor Recirculation Pump Motor/Generator (MG) Set drive motors. |
| 08:11:50 | The RCIC injection valve begins to open. |
| 08:11:51 | Outboard Drywell Equipment Drain Isolation Valve 2-G16-F020 is closed. Closure time exceeds expected time (11 vs. 9 seconds) per PT-11.4 Acceptance Criteria. |

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# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

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## ATTACHMENT 1 (CONT.)

| | |
|---|---|
| 08:11:54 | HPCI Turbine Stop Valve begins to open. The turbine begins to increase to rated speed. The HPCI injection valve does not open, as reactor level is above the initiation setpoint when the "valve open" permissives are satisfied. HPCI, per design, will not inject for approximately 13 seconds following initiation due to the permissives which must be satisfied to open the injection valve. |
| 08:12:04 | RCIC system reaches rated injection flow. |
| 08:12:14 | ERFIS computer printout shows momentary opening of Safety Relief Valve (SRV) B21-F013K. This is not an accurate accounting of the valve actuating. Valve setpoint was never reached, and no other indication showed the valve opening. The computer had been printing out this indication prior to the event start. |
| 08:13:05 | Operator manually opens the HPCI injection valve to assist RCIC in restoring level. |
| 08:13:20 | HPCI exceeds rated injection flow. |
| 08:14:27-35 | Reactor water level has been restored above the low level SCRAM setpoint. The four low level subchannels reset at this time. |
| 08:15:08 | Operator observes that Reactor water level has been restored above the low level SCRAM setpoint and manually initiates closure of the HPCI injection valve. |
| 08:15:33 | Reactor water level goes above the low level alarm setpoint. |
| 08:16:06 | Operator manually opens HPCI discharge path to the condensate storage tank (CST). This places HPCI in the Pressure Control mode. |
| 08:16:51 | Hi Reactor vessel level alarm setpoint is reached (192"). |
| 08:17 | Operator secures pressure control mode of HPCI by closing the discharge to CST. |
| 08:18 | Operator manually trips the HPCI turbine. |
| | Operator opens the RWCU reject valve to establish a method of level control. |
| 08:20 | Operator closes the RCIC injection valve. |
| 08:22 | Operator resets the group isolation signals. |

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ATTACHMENT 1 (CONT.)

| | |
|---|---|
| 08:24 | Operator secures the RCIC turbine and the "A" and "C" heater drain pumps. |
| 08:28 | Hi Reactor level alarm resets. |
| 08:29 | Operator starts the 2A RFP and places the Startup Level Control Valve (SULCV) in service. |
| 08:30 | Rod position scan performed; all rods confirmed to be fully inserted. |
| 08:40 | Operator attempts to start the 2A Reactor Recirculation pump. The M/G set drive motor breaker closes, then immediately re-opens. |
| 08:45 | Operator attempts to start the 2B Reactor Recirculation pump. The M/G set starts, but the field breaker does not close. The drive motor breaker is then re-opened. |
| 08:47 | Alternate Rod Injection is reset. |
| | Automatic function of SCRAM channels "A" and "B" are reset. |
| | Manual SCRAM Channels "A" and "B" are reset. |
| 08:49 | Operator momentarily places the mode switch in "REFUEL". Operator realizes error, and immediately returns the switch to "SHUTDOWN". Manual SCRAM signal is received on both channels. |
| 08:50 | Operator resets the manual SCRAM signals. |
| 08:51 | Attempt made to open B32-F031B, the Reactor Recirculation pump discharge valve. Valve stopped with intermediate position indication. Thermal trip reset, but valve would not continue open. Work Request/Job Order (WR/JO) 91-ABRF1 initiated. |
| 09:01 | HPCI Auxiliary Oil pump is secured, and HPCI restoration to standby alignment is begun. |
| 09:07 | Drywell Floor and Equipment Drain Inboard and Outboard Isolation valves are opened. |
| 10:24 | RFP 2A is secured in accordance with GP-05. Remainder of shutdown proceeded in accordance with GP-05. |