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MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

THRU: Karl V. Seyfrit, Chief
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Office for Analysis and Evaluation
of Operational Data

FROM: Thomas R. Wolf
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SUBJECT: ENGINEERING EVALUATION REPORT - BRUNSWICK STEAM ELECTRIC PLANT UNIT 2 LOSS OF RESIDUAL HEAT REMOVAL SERVICE WATER - JANUARY 16, 1982

Enclosed is my evaluation of a loss of residual heat removal service water event which occurred at Brunswick on January 16, 1982. My principle conclusions are that this event is of minor significance and that it should be not classified as an abnormal occurrence.

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Enclosure:
As stated

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BRUNSWICK NUCLEAR POWER STATION UNIT 2
LOSS OF RESIDUAL HEAT REMOVAL
SERVICE WATER ON
JANUARY 16, 1982

ENGINEERING EVALUATION REPORT

by the

REACTOR OPERATIONS ANALYSIS BRANCH

OFFICE FOR ANALYSIS AND EVALUATION

OF OPERATIONAL DATA

August 1982

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Prepared by: Thomas R. Wolf

BRUNSWICK UNIT 2

LOSS OF SERVICE WATER BOOSTER PUMPS

SUPPLYING THE RESIDUAL HEAT REMOVAL

HEAT EXCHANGERS

1.0 BACKGROUND

Carolina Power Light (CP&L), as owner and operator of the Brunswick Steam Plant (BSEP) Unit 2, reported in Licensee Event Report (LER) 82-005/01T that on January 16, 1982 an unsuccessful attempt was made to initiate normal suppression pool cooling via the residual heat removal (RHR) system.[1] This try came following a sequence of occurrences which included a main turbine trip, a reactor scram, a loss of normal feedwater, and a reactor core isolation cooling (RCIC) system initiation. Normal suppression pool cooling (as well as normal shutdown cooling) could not be attained because both residual heat removal service water (RHRSW) trains were inoperable. These RHRSW trains were declared inoperable when none of the four SW booster pumps to the RHR heat exchangers (RHRHX) could be started. Normal cooling of the reactor by utilizing main feedwater steaming to the condenser was restored within a half-hour of the sequence initiation. After maintenance and testing, RHRSW "B" train was declared operational within 4.25 hours of the sequence start and "A" train and "A" train within 8 hours. At no time during this event were any safety limits exceeded.

This report documents an AEOD review of this event. It is based on information included in the Licensee Event Report, NRC regional and resident reports, licens responses to these reports and personal telephone conferences and meetings betwe the author and the licensee.

2.0 CHRONOLOGY

Based on site visit and telecon discussions with BSEP personnel and the NRC Region II special safety inspection report, the following chronology of the January 16, 1982 loss of RHRSW event was developed.[2,4,5]

SEQUENCE OF OCCURRENCES

Time approximate	Description
<1625	Reactor power @100%; Pressure sensing instrumentation in steam jet ejectors (SJAE) develops trouble; Condenser vacuum decreases; Power reduction initiated.

One set of SJAE lost due to inadvertent short to ground of a hot lead by electrical maintenance technicians as they were replacing a SJAE pressure sensing instrument.

Select rod insert commanded; Recirculation pumps speed reduced to reduce recirculation flow; Mechanical vacuum pump started; SJAE restart attempted.

Reactor power @30-40%; Low condenser vacuum; Turbine stop valve fast closure; Reactor scram.

Group 1 isolation (main steam isolation valves [MSIV] close); Isolation occurred when the mode switch was switched from RUN to the SHUTDOWN position by normal procedure. Steam flow switches apparently still indicated greater than 40% flow, which initiates a Group 1 isolation with the lost due to loss of pump driving steam on MSIV closure.

RCIC manually started with suction from condensate storage tank; Suppression pool temperature @73° - 74°F; Drive steam to RCIC turbine maintains reactor coolant system (RCS) pressure; Per plant procedures, operator attempts to initiate RHR suppressio pool cooling by starting "B" train of RHRSW; RHRSW "B" train booster pumps suction header pressure switch PS-1176 low pressure alarm (<20 psi); RHR "B" train booster pumps (2B and 2D) prevented from starting by low suction pressure interlock; Operator attempts to start "A" train of RHRSW; RHRSW "A" train booster pumps suction header pressure switch PS-1175 low pressure alarm (<20 psi); RHRSW "A" train booster pumps (2A and 2C) prevente from starting by low suction pressure interl Control panel booster pump suction pressure indicated @60 psi; RHRSW declared inoperable Maintenance request initiated.

Group 1 isolation signal reset; MSIV reopene Condenser vacuum restored.

1655	Reactor feed pump started re-establishing feedwater flow; RCIC secured; Suppression pool temperature @75° - 76°F.
1710	Technician discovers PS-1176 power feed 120v-ac breaker open; breaker manually closed; RHRSW "B" train booster pump interlock automatically clears; RHRSW "B" train booster pumps started and associated RHR train aligned and operated in suppression pool cooling mode.
1810-2058	RHRSW "B" train cycled on and off several times to run further operability tests.
2058	RHRSW "B" train declared operational.
2354	After maintenance and testing, RHRSW "A" train declared operational. (PS-1175 repaired. Failure due to leakage of operating fluid in diaphragm housing.)

3.0 FAILURE MECHANISMS/CAUSES/CORRECTIVE ACTIONS

3.1 BASIC MECHANISMS

RHRSW booster pumps suction pressure is sensed by two Barksdale pressure switches, one per booster pump train. Each switch is utilized in the RHRSW booster pump control logic in such a manner that if a low suction pressure is indicated by a switch, both booster pumps in the associated train are prevented from either starting and running or continuing to run. In this event, it was found that the "B" train pressure switch (2-SW-PS-1176) was inoperable due to a circuit breaker (circuit breaker 19 on panel 2B located in the cable spreading room) being open. This interrupted power to the low suction pressure protection logic circuit causing an electrical start inhibit of pumps 2B and 2D.

The "A" train pressure switch (2-SW-PS-1175) was found to be inoperable due to air accumulation in the oil-filled chemical seal attached to the pressure switch. To prevent chemical corrosion of the Barksdale pressure switch, the pressure switch is isolated from the brackish SW by a diaphragm and a short section of pipe filled with glycerol, which contacts the pressure switch. Technicians found that the oil had apparently leaked from the chemical seal, allowing an air bubble to form which render the pressure switch inoperable and resulted in the generation of a start inhibit of pumps 2A and 2C.

3.2 CAUSES/CORRECTIVE ACTIONS

Circuit Breakers

Circuit breaker 19 was apparently incorrectly left open or it spuriously tripped open following a well water flush of the RHRSW piping conducted earlier that day An entry in a facility log shows that the flush operation was completed at 4:50 on January 16. The procedure OP-43 Service Water System, Revision 20 approved September 30, 1981, specifies flushing the RHRSW piping with fresh water to remo salt water to prevent corrosion of the pipe. Step G.3.2 of the procedure specif that breaker 19 be opened during flushing operations to allow the motor cooler supply solenoid valve to open to permit the motor coolers to be flushed along wi the rest of the piping. Circuit 19 supplies power to operate the motor cooler supply valves for pumps 2B and 2D and the valves fail open on a loss of power; therefore, opening the appropriate circuit breaker is a convenient way to open the valve for the flush. Step G.3.10 of the procedure specifies that the breake be reclosed after the flush.

Procedure OP-43 does not require that individuals performing step G.3.10 sign off or otherwise indicate that the step was completed. However, in their letter of December 31, 1980, in response to NUREG 0737 Clarification of TMI Action Plan Requirements, CP&L committed to the following in regard to item I.C.6 (Verify correct performance of operating activities).

> "When returning equipment to service which has not been under clearance, for example, instruments or hydraulic snubbers removed for surveillance testing, a second person will verify proper system alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are currently aligned. The person performing the verification will have the qualifications necessary for returning the equipment to service or will be a QA inspector".

On July 10, 1981 the NRC issued an order to CP&L which required implementation by January 1, 1982 of procedures to verify correct performance of operating activities as specified by NUREG 0737 item I.C.6.

During a special inspection following this event, an NRC inspector observed that in general, valve lineups have requirements for double verification, but certain other procedures, such as periodic test procedures for surveillance testing of technical specification required equipment, do not require double verification.

Procedure OP-43 is, however, an example in which this commitment to double verification was not implemented. Historically, several other events have occurred at BSEP that served as precursors to this event involving circuit breaker No. 19 being incorrectly positioned. Three were reported to the NRC in Licensee Event Reports (LERs) as described below.

LER 1-80-8: On January 15, 1980 circuit 19 was found deenergized on Unit 1. The licensee concluded that the breaker was left open as a result of a system flush 11 hours earlier. To prevent recurrence, a precaution was placed in procedure OP-43.G to advise the operators that the RHRSW pumps are inoperable with the breakers open.

LER 2-80-66: On September 5, 1980 during normal operation, the corresponding breaker for the Unit 2 A loop tripped, rendering the 2A and 2C pumps inoperable. No cause could be found so the breaker was reset and operation continued.

LER 1-81-95: On December 6, 1981, the Unit 1 B loop of RHRSW was being put in service for torus cooling when it was identified that the pumps would not start. The cause was identified as breaker 19 being open. An investigation by the licensee failed to identify any individual who would admit being responsible for opening the breaker. As corrective action, the breaker panels were locked closed.

After the January 16, 1982 event, the subject circuit breakers and associated circuits were functionally tested but no abnormalities were revealed. As a precaution, the breaker was replaced with an identical unit. It was determined, however, that the particular circuit breaker involved is designed such that it cannot be readily determined upon visual inspection whether the breaker is in the tripped or untripped position.

Because of the lack of proper system alignment verification and the problem with the circuit breaker position indication, it is uncertain if step G.3.10 was eithe omitted entirely on the morning of January 16, attempted but not completed correctly, or a spurious breaker trip occurred after step G.3.10 was completed.

In response to a Notice of Violation issued by the NRC as a result of the special inspection following this event, the licensee stated that an auxiliary operator (AO) other than the AO assigned to perform the RHRSW flush per OP-43 was requested to open circuit breaker 19.[3] The AO performing the flush did not check to see if breaker 19 was closed after the flush and the control operator did not follow up on the flush to assure its proper completion. A new procedure has been written to provide adequate guidance for performing the flush. Included in this new procedure is the requirement for a double verification signoff to help assure that the breakers (circuit breakers 19 for train B and 22 for train A) are correctly positioned. The licensee has also established a program to ensure that all independent verification steps are being accomplished and are in total compliance with previous commitments to the NRC. In addition, since the breaker involved is of a type commonly used in the plant, consideration is being given to replace all such breakers with ones giving more noticeable position indication. From a human factors viewpoint, such a modification would improve the operational as well as safety security of the affected systems.

Pressure Switches

To recover from the January 16 event, the seal chamber of PS-1175 was refilled with oil and the switch was recalibrated. During the post-event special inspection, and the NRC inspector reviewed calibration records for PS-1175 and PS-

1176 and found that numerous problems had occurred with the switches. Records show that switch 1175 was found to be inoperable on August 4 and October 31, 1981, due to an oil leak from the chemical seal. On November 2, 1981 a new switch was installed. Switch 1176 was also found inoperable on March 20 and September 3, 1980, with no logged entry of the exact cause. On July 7, 1981 the switch was replaced. On July 11, 1981 the switch was found to again be inoperabl due to an oil leak.

Calibration records demonstrate that these switches as installed and/or maintaine are unreliable. The pressure tap comes off piping in the overhead and runs down to the instrument. Maintenance personnel state that this causes a recurrent problem of plugging of the instrument lines with debris. As initially reported in Licensee Event Report (LER) 2-82-005/01T, the loss of the pressure switches in this event was attributed to sensing line fouling. In subsequent discussions with the licensee, however, they stated that instrument line plugging was not a factor in this event and had never been a problem. Searches of the available data also do not support the contention that plugging is a recurrent problem.

Procedure MI-3-3A34, S.W. - P.S. 1175 and 1176 Service Water Pressure Switch, D2T-M8055-L6, revision 0 December 26, 1979 is the specific procedure intended to be used to calibrate pressure switches 1175 and 1176. The frequency of calibration is listed as semiannual. Calibration records from September 1979 to date were reviewed by an NRC inspector for both units and several discrepancies were found.

With respect to PS 1175 and 1176, Procedure MI3-3A34 specifies, that this is not a Q-list (safety-related) item and that this procedure is not technical specification related. The procedure is in conflict with Volume XI, Book 2, Table I of the Brunswick Plant Operating Manual, Q-list which correctly identified these two switches as Q list items.

Volume XI, Book 2 of the Plant Operating Manual contains the plant Q-list in several parts. Table I in Book 2 lists Q-list items on a plant system basis and the service water system section lists SW-PS-1175 and 1176 as Q list items. Table I.a of that procedure is a computer output listing of Q list instruments titled Brunswick Instrument Calibration Cross Reference, Revision 11 May 25, 1979. Table I.a does not appear to list the subject switches and thus there is an apparent discrepancy between the two lists. Table I.a is often used by plant personnel for quick reference to determine if a particular instrument is on the Q-list and therefore it should be current, complete and accurate.

PS-1176 for Unit 2, as found by the NRC inspector, has no record of calibration between September 2, 1980, and July 11, 1981. This interval exceeds the six month frequency specified. Although approved on December 26, 1979, procedure MI3-3A34 was apparently not fully implemented. The majority of the completed data sheets are not the one from the approved procedure; rather they come from procedure MI3-3A34, Procedure for General Calibration of Pressure Switches. The frequency for MI3-3A34 states "As Required" and this procedure is stated to be used for non Q-list pressure switches. Records show that of the 24 calibrations performed since December 26, 1979, on these pressure switches on both units, only five used procedure MI3-3A34 and the rest used procedure MI3-3A. Some used procedure MI3-3A but the data was recorded on a data sheet from procedure MI3-3 dated February 12, 1976. Calibration data recorded was adequate, however, with the exception that data sheet 3-3A34 requires a signature reflecting that permission was granted by the Shift Foreman to remove the instrument from service.

In response to the special investigation report finding which detailed these problems, the licensee has acknowledged the shortcomings and has initiated corrective actions. The table (I and IA) in Volume XI have been revised to assure that all Q-list equipment is correctly identified on both tables. Instructions for retrieving a correct maintenance instruction for a particular instrument have been provided at each maintenance computer console. All Q-list instrumentation has been entered into the Periodic Maintenance Scheduling Progran to assure a proper calibration schedule. PS-1175 and PS-1176 have been placed or a monthly calibration schedule until either a more reliable history has been established, the switches are replaced with more reliable switches, or the switch are removed from the RHRSW pump logic. An engineering work request (EWR) has bee issued to Plant Engineering to devise a permanent solution to identify and correc these switches. Also, future maintenance on these switches will consist of a total switch replacement instead of an individual component replacement to assure that the diaphragm fluid seal is properly in place before a switch is returned to service following maintenance. This change out procedure has been tested and takes approximately 20 minutes. If for some reason change out cannot be done, such as lack of a replacement switch, procedures have been developed to jumper ou the interlock. This task takes about five minutes. Finally, a Work Order Tracki System (WOTS) was put into operation on January 1, 1982. WOTS will allow Mainten personnel to readily access post-maintenance work orders to allow early recogniti of repetitive failures.

Booster Pump Interlock Logic

While each train of the RHRSW booster pump system contains redundant pumps, the utilization of only one pressure switch per train to control the start interlock on both pumps compromises this redundancy and makes it susceptible to single failures. To obviate this situation, consideration is being given to modify the logic by:

a. Adding a redundant pressure switch in each loop;

b. Adding suction valve position indication into the circuitry; (The contro logic diagram in the NRC document files shows this logic but discussions with the licensee indicate that this logic was in the original design but only the pressure switch portion was finally implemented.) or

c. Deleting the suction pressure interlock (as noted previously) but replac ing it with a valve position interlock.

4.0 EVALUATION/CONCLUSIONS/RECOMMENDATIONS

To control suppression pool temperature, BSEP procedures require the start of suppression pool cooling whenever a potential heat input source develops no matter what the heat source cause or pool temperature. Thus, starting of suppression po cooling upon RCIC initiation (heat source being RCIC pump turbine drive steam exhaust) is a normal procedure and not necessarily indicative of any unusual or serious problem. In the January 16 BSEP event, the loss of SJAE and subsequent loss of the normal primary heat sink systems, i.e., feedwater and condenser, was not too significant. To maintain the reactor vessel water level and while continu to remove the core decay heat, the RCIC was started, an expected operation under these conditions. The operation of the RCIC was sufficient to keep the core parameters stable by injection of condensate storage tank water into the vessel and steam generated and released to the suppression pool via the RCIC turbine pump drive system. Since steam was being dumped into the suppression pool, per plant procedures, the control room operators attempted to align and start the RHR system in the suppression pool cooling mode even though the technical specification temperature limits of the suppression pool were not even close to being violated. Everything was proceeding normally with no safety concerns until the RHRSW booster pumps could not be started which, consequently per operational guidelines, necessitated the declaration ofthe RHRSW system as inoperable event though the main SW pumps were not affected.

Since the suppression pool temperature was well within limits, the failure to start of the booster pumps was not immediately significant except for the fact that identical portions of both trains of a safety system were inoperable at the same time due to unknown causes. BSEP operational guidelines address the case where normal RHRSW is lost and RHR cooling in any mode is required. Several alternatives are given depending on the particular circumstances. These include:

a. Supply the RHRHX from the SW system utilizing only the main SW pumps without the use of the RHRSW booster pumps. Water can be supplied to the RHRHX in sufficient quantities to meet all heat removal requirements via this method; however, the SW-to-reactor water positive differential pressure across the RHRHX for radioactive fluid outleakage control will be lost.

b. Utilize available manual connections between the SW and the fire protection system. The fire protection pumps develop sufficient head and flow rates to replace the SW but this source is limited by its water supply storage capacity. The fire protection storage supply consists of 200,000 gallon minimum technical specification volume in a dedicated 300,000 gallon capacity tank and a connection to the 90,000 gallon minimum technical specification volume in the 150,000 gallon capacity demineralized water storage tank.

c. At low RHR heat removal rates, utilize available RHR connections to the spent fuel cooling heat exchangers.

Service water system cross ~~connects~~ between units are not included in the BSEP system design. Therefore, at BSEP, this potential cooling method is unavailable. While all of the alternatives were available, none had to be utilized because the normal heat removal systems, i.e., main feedwater and condenser, were returned to service and the RCIC secured before any technical specification suppression pool limits were reached.

Upon consideration of all of these factors, the significance of this event is minor. Total loss of RHRSW did not occur, only the booster pumps/RHRHX differential pressure feature was primarily affected. No common mode or common cause failures were demonstrated. The causes, however, were considered to be significant enough for the NRC to cite the licensee for four violations, three of severity level IV and one of severity level V. The licensee has responded to these citations in a favorable manner with appropriate NRC verification programs being undertaken.

To improve the system operability and reduce the significance of single failures on the operation of the booster pumps, the licensee is studying the system contro logic. This topic seems to be adequately addressed Study conclusions should be all reactors having similar pumping systems.

The licensee investigations of the circuit breaker and pressure switch designs als seem to adequately address the problems manifested of the components by this event Study progress and results along with generic considerations should be monitored by appropriate industry and NRC personnel.

5.0 ABNORMAL OCCURRENCES CLASSIFICATION

An NRC policy statement published in the Federal Register (42 FR 10950) on February 24, 1977, sets forth the classification criteria for an event as:

> "An event will be considered an abnormal occurrence if it involves a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety could include but need not be limited to: (1) Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission; (2) Major degradation of essential safety-related equipment; or (3) Major deficiencies in design, construction, use of, or management controls for licensed facilities or material."

For commercial nuclear power plants this policy statement notes that examples of events which might qualify under this criteria include:

A. Malfunction of Facilities, Structures or Equipment:

1. Exceeding a safety limit of licensee technical specifications (10 CFR 50.36d(c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

B. Design or Safety Analysis Deficiency, Personnel Error, or Procedural or Administrative Inadequacy:

1. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or technical specifications that require immediate remedial action.

2. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

The January 16, 1982 event at BSEP involved the loss of both trains of RHRSW booster pumps. This loss was a result of design problems (pump interlock system, circuit breaker positive position indication and, possibly pressure switch seal system), personnel errors (circuit breaker position and pressure switch calibration and) procedural deficiencies (circuit breaker position check and pressure switch calibration). Since these problems combined only affected the booster pumps and not the main SW pumps, SW to the RHRHX was available throughout the event. Consequently, no degradation of the fuel, primary coolant pressure boundary and containment and no safety limits were exceeded.

Considering all of these items, it does not appear that the circumstances encountered in this event are necessary and sufficient to classify this event as an abnormal occurrence. Thus, it is concluded that this event is <u>not</u> an abnormal occurrence.

REFERENCES:

(1) Letter, C. R. Dietz, BSEP, to J. P. O'Reilly, NRC, enclosing Brunswick Steam Electric Plant Licensee Event Report 2-82-5, dated January 29, 1982.

(2) Letter, R. C. Lewis, NRC, to J. A. Jones, CP&L, "Reports Nos. 50-324/82-10 and 50-325/82-10," dated April 2, 1982.

(3) Letter, B. J. Furr, CP&L, to J. P. O'Reilly, NRC, enclosing Brunswick Steam Electric Plant "Response to Infractions of NRC requirements," dated May 24, 1982.

(4) Memorandum, T. R. Wolf to S. Rubin, "Telecon Notes - Conversation with Carolina Power and Light Personnel Concerning January 16, 1982 Loss of Residual Heat Removal Service Water Event at Brunswick Steam Electric Plant - LER 2-82-005/01T," dated June 4, 1982.

(5) Memorandum, M. El-Zeftawy and T. R. Wolf to C. Michelson, "Site Visit/ Meeting Notes - Brunswick Steam Electric Plant - March 24, 1982," dated July 6, 1982.