Progress Energy

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

LICENSEE EVENT REPORT NO. 2003-002-00
FAILURE OF AUTOMATIC CONTAINMENT
VENTILATION ISOLATION DURING CONTAINMENT PRESSURE RELIEF

Ladies and Gentlemen:

The attached Licensee Event Report is submitted in accordance with the requirements of 10 CFR 50.73. Should you have any questions regarding this matter, please contact Mr. C. T. Baucom.

Sincerely,

Timothy P. Cleary

Timothy P. Cleary
Plant General Manager

CAC/cac

Attachment

c: Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspector, HBRSEP

**Progress Energy Carolinas, Inc.**
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| NRC FORM 366 (7-2001) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004 |
|---|---|---|
| **LICENSEE EVENT REPORT (LER)** (See reverse for required number of digits/characters for each block) | | Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection |

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TITLE (4)

Failure of Automatic Containment Ventilation Isolation During Containment Pressure Relief

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MO | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MO | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 06 | 05 | 2003 | 2003 | 002 | 00 | 07 | 31 | 2003 | FACILITY NAME | DOCKET NUMBER |

| OPERATING MODE (9) | 1 |
|---|---|
| POWER LEVEL (10) | 100% |

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR ' : (Check all that apply) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(ix)(A) |
| | 20.2201(d) | | 20.2203(a)(4) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(1) | | 50.36(c)(1)(i)(A) | | 50.73(a)(2)(iv)(A) | | 73.71(a)(4) |
| | 20.2203(a)(2)(i) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(v)(A) | | 73.71(a)(5) |
| | 20.2203(a)(2)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(B) | | OTHER Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iii) | | 50.46(a)(3)(ii) | X | 50.73(a)(2)(v)(C) | | |
| | 20.2203(a)(2)(iv) | | 50.73(a)(2)(i)(A) | X | 50.73(a)(2)(v)(D) | | |
| | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(vii) | | |
| | 20.2203(a)(2)(vi) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(viii)(A) | | |
| | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(B) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| C. T. Baucom | 843-857-1253 |

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

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| B | JM | HS | GEMCO | Y | | | | | |

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


**ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

At approximately 19:49 hours on June 5, 2003, during a planned source check of containment radiation monitor R-11 with a routine containment pressure relief in progress, the containment pressure relief isolation valves would not close automatically. The valves were closed by use of the control switch in the control room to stop the pressure relief of the containment at the time of the automatic isolation failure. The source check should have caused the valves to close automatically by the initiation of a containment ventilation isolation signal. The penetration was isolated at 20:47 hours by the use of a closed and de-activated automatic isolation valve. The cause of the failure was determined be a failure of the control switch that controls the valves for the containment pressure relief penetration. The control switch was repaired at approximately 14:13 hours on June 6, 2003, and the system was restored to operable status at that time. A planned and monitored gaseous release from the containment was in progress at the time of this event using the containment pressure relief system. No release limits were exceeded. If plant conditions had required isolation of the penetration, alarms and indications in the control room would have alerted the operators to the condition and the applicable operating procedures direct the operators to manually isolate the penetration. Investigation of the event determined that an inappropriate design change occurred in about 1980, which when coupled with the switch failure, caused the loss of safety function. Therefore, this event is being reported in accordance with 10 CFR 50.73(a)(2)(v)(C) and (D).

# LICENSEE EVENT REPORT (LER)
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**NARRATIVE** *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

I. DESCRIPTION OF EVENT

At approximately 19:49 hours on June 5, 2003, during a planned source check of containment radiation monitor R-11 [IL,IK:RI] with a routine containment pressure relief in progress, the containment pressure relief isolation valves, V12-10 and V12-11 [VA,VL:ISV], would not close automatically. The valves were closed by use of the control switch [JM:HS] in the control room to stop the pressure relief of the containment at the time of the source check automatic isolation failure. The source check should have caused the valves to close automatically by the initiation of a containment ventilation isolation signal. The penetration was isolated at 20:47 hours by the use of a closed and de-activated automatic isolation valve; specifically, valve V12-10 was a closed and de-activated, in accordance with Technical Specifications Limiting Conditions for Operation (LCO) 3.6.3, Containment Isolation Valves, Required Action B.1.

The cause of the failure was determined be a failure of the control switch that controls the valves for the containment pressure relief penetration. The control switch was repaired at approximately 14:13 hours on June 6, 2003, and the system was restored to operable status at that time. A planned and monitored gaseous release from the containment was in progress at the time of this event using the containment pressure relief system. No release limits were exceeded. If plant conditions had required isolation of the penetration, alarms and indications in the control room would have alerted the operators to the condition and the applicable operating procedures direct the operators to manually isolate the penetration.

A root cause investigation was completed for this event. The results of that investigation were reviewed by the Plant Nuclear Safety Committee (PNSC) on July 9, 2003, in accordance with the Updated Final Safety Analysis Report (UFSAR) Chapter 17 requirement for the PNSC review of reportable events. That investigation concluded that the switch failure caused the event, but also that the design of this containment isolation circuitry had apparently been unintentionally modified, in about 1980, such that the switch failure alone could have prevented the automatic isolation of this penetration.

The Containment Pressure and Vacuum Relief System [VA,VL] is provided to control variations in containment pressure with respect to atmospheric pressure. These variations are due to changes in atmospheric pressure and leakage in containment

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**NARRATIVE** *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

from the Instrument Air and Penetration Pressurization Systems. The Containment Pressure and Vacuum Relief System includes separate 6 inch lines penetrating the containment [NH], each equipped with two quick-closing, tight-seating, 125 psi air operated butterfly valves, one inside and one outside containment. These valves are designed to fail closed on loss of control signal or control air, and are closed during normal plant operation, except as required for containment pressure control.

The butterfly valves are protected by debris screens, located inside containment and attached to the inboard pressure and vacuum relief valves, which will ensure that airborne debris will not interfere with their tight closure. The pressure relief line discharges to the plant vent through a High-Efficiency Particulate Air (HEPA) filter and charcoal filters [VL:FLT]. These filters are provided for removal of iodine and particulate radioactivity from the vented air. Operation of the pressure and vacuum relief lines is manually controlled by the plant operator. A narrow range pressure transmitter [IK:PIT] continuously indicates containment pressure in the control room. Separate high and low pressure alarms [IK:PA] are actuated by this transmitter to alert the operator to overpressure and vacuum conditions. Vacuum relief can be accomplished without regard to atmospheric conditions. In the event of pressure buildup, the operator will be guided by atmospheric conditions, and by the containment particulate and radio gas monitor [IK,IL:RI] in relieving the overpressure. Manual operation of both these lines is overridden by automatic containment ventilation isolation and containment high radioactivity signals [JM].

Containment pressure relief isolation valves V12-10 and V12-11 are designed with a common control switch located on the Reactor-Turbine Gauge Board (RTGB). The V12-10 and V12-11 RTGB control switch has “Open” and “Close” switch positions and a spring return to center mechanism. The valve circuitry is also designed with position indication available in the control room on the RTGB.

The valve circuitry is designed such that the air-solenoid valves [LD,VA,VL:ISV] for V12-10 and V12-11 are energized to open the valves when the operator places the control switch in the “Open” position. The control circuit for these valves was inadvertently modified, in about 1980, such that the “Open” switch contact was electrically paralleled with the automatic containment isolation circuit functions and the close function of the control switch. Therefore, when the “Open” switch contact remained closed due to the control switch failure, both air operated solenoid valves remained continually energized, keeping the valves open. The

**NARRATIVE** *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

presence of an automatic isolation signal was not able to electrically interrupt the continually energized path to the solenoid valves.

A letter to the NRC dated March 11, 1980, in response to Three Mile Island short term lessons learned requirements, stated, "The second category of switch is that which uses a single set of contacts for both trains. Normally, this would be unacceptable as the switch would not be single failure proof. However, when the switch is wired is series with the Train 'A' and Train 'B' containment isolation relay contacts, the logic system which controls the isolation valve is single failure proof and, therefore acceptable." As stated previously, a modification in the 1980 time-frame inadvertently circumvented this design requirement.

The automatic isolation function is tested in accordance with Technical Specifications (TS) Surveillance Requirement (SR) 3.6.3.5, with a required frequency of 18 months, corresponding to the periodicity of scheduled refueling outages. This SR, which is conducted by Operations Surveillance Test (OST)-163, was successfully completed during the last refueling outage in November 2002. Therefore, it is likely that the switch failure occurred after that time, although the specific time and date of failure is not known. The switch failure was discovered on June 5, 2003, due to the performance of the containment radiation monitor source check coincidentally with a containment pressure relief.

A review of containment penetrations was conducted to determine if another set (inboard and outboard) of containment isolation valves are designed similar to the V12-10 and V12-11 control circuit. The results of the review indicate that the control circuitry for the containment vacuum relief isolation valves, V12-12 and V12-13, were modified by the same modification in about 1980 and the same design deficiency was also introduced into that control circuitry.

Plant control circuits that utilize the same GEMCO Switch Model (404S-3-2-1-2-2-Y-EE2) were reviewed to determine the impact of switch failure. The acceptance criteria applied in the review was that the RTGB control switch failure did not adversely affect accident mitigation. The results of this review concluded that the function of the switch in other control circuits is either non-safety-related, or if safety-related, is bounded by the single failure analysis for that system.

**NARRATIVE** *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

A review of radiation monitor control circuits concluded that no single switch controls multiple devices and that no switch failure could disable the associated automatic actuation functions.

A review of the documentation associated with the modifications to the containment pressure and vacuum relief control circuits in the 1980 timeframe concluded that the modification design failed to identify the applicable design requirement pertaining to the single control switch. At the time of the modification, engineering procedures were less detailed and the design basis requirements were more difficult to identify and retrieve. Since that time, design basis documents for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, have been created, and the testing and design guidance provided in the engineering procedures have been improved. Therefore, it is concluded that the current design barriers effectively prevent recurrence of this type of modification error. This, coupled with the planned and completed circuitry design reviews, provides assurance that other errors of this type are not being introduced.

## II. CAUSE OF EVENT

The cause of the event was determined to be the control switch failure coupled with the design deficiency of this containment isolation circuitry that had apparently been unintentionally modified, in about 1980, in a manner such that the switch failure alone could have prevented the automatic isolation of this penetration.

## III. ANALYSIS OF EVENT

LCO 3.6.3, Required Action B.1, establishes the appropriate compensating action and completion time for inoperability of two containment isolation valves in a penetration. Required Action B.1 requires the penetration to be isolated by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange, within 1 hour. As stated in the event description, the penetration was isolated within 1 hour by the use of a closed and de-activated automatic isolation valve; specifically, valve V12-10 was closed and de-activated.

Valves V12-10 and V12-11 are normally closed. These valves are opened periodically to allow containment pressure relief, and are designed to close automatically on receipt of a containment isolation or containment high radiation signal. If an accident had occurred resulting in release of radioactivity into containment during the time that

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the control switch for the valves was malfunctioning, and if a containment pressure relief had been occurring at that time, it is possible that unavailability of the automatic closure function could have delayed or prevented valves V12-10 and V12-11 from automatically closing. This could have resulted in an unintended, although monitored, release.

For the purpose of assessing the risk impact of this event, it was assumed that the condition may have existed since the last refueling outage. This is approximately 28 weeks, and estimating approximately 7.4 hours of pressure relief per week results in a rough estimate of 208 hours of time when containment pressure relief operations were taking place. The average annual effect of the condition on Large Early Release Frequency (LERF) may therefore be estimated by assuming the risk existed for a fraction of the time equal to 208 / 8760 or 0.024.

A calculation was performed to estimate the likelihood of the failure of operators to manually position the control switch to isolate V12-10 and V12-11. Based on the assumptions that there would be one half hour of available time before a large fraction of the containment volume could be released through these valves, that the action would normally be accomplished within 5 minutes of indications that the action is needed, and that the action requires only 1 minute to accomplish, a probability of failure of 0.001 was estimated for this CP1 action, i.e., “an action to be taken in response to direct cue.”

The probabilistic safety assessment analysis currently estimates a core damage frequency of 4.32E-5 / year. A fraction of this core damage frequency is associated with accident sequences which are expected to result in early failure or bypass of the containment. Contributions from these sequences would be unaffected by postulated failures of V12-10 and V12-11, and their contributions to core damage frequency could be subtracted before evaluating the consequences of the switch malfunction. For simplicity, this correction was not made. This simplification results in a conservative estimate of the risk.

Therefore, the increase in large early release frequency associated with this event can be calculated: 4.32 E-5 / year * 0.024 * 0.001 = 1E-9 /year.

This indicates that the risk associated with the control switch malfunction was negligible.

As stated previously, no release limits were exceeded. If plant conditions had required isolation of the penetration, alarms and indications in the control room would have alerted the operators to the condition and the applicable operating procedures direct the operators to manually isolate the penetration. This event is being reported in accordance with 10 CFR 50.73(a)(2)(v)(C) and (D), any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to control the release of radioactive material or mitigate the consequences of an accident.

IV. CORRECTIVE ACTIONS

The corrective actions for this event that have been planned or completed include the following:

- Procedures used to operate the Containment Pressure and Vacuum Relief System have been revised. The revised procedures require isolation of the penetration by use of a closed and deactived automatic valve, and subsequent verification that the switch is not in a failed state such that it would prevent the automatic isolation function prior to use of the system for containment pressure or vacuum relief. After verification of switch operability, the closed and deactivated valve is restored and the system is used to accomplish the needed function of containment pressure or vacuum relief.

- The defective switch has been replaced.

- A temporary modification will be implemented, which will restore the design basis requirement that the switch failure will not prevent the safety function. This is a planned action with a scheduled completion date of September 30, 2003.

- The control switch for the containment vacuum relief isolation valves (V12-12 and V12-13) will be replaced and the removed switch will be evaluated for any signs of degradation. This is a planned action with a scheduled completion date of October 25, 2003.

- A review of the safety-related component control wiring diagrams will be performed to verify that redundant components are designed with individual control switches in each circuit, and that failure of individual component control switches alone will not cause a failure to fulfill a safety function. This is a planned action with a scheduled completion date of October 25, 2003.

– It will be confirmed that functional testing verifies operability of the isolation function or procedural guidance will be established that verifies operability of the isolation function when a source is applied to containment radiation monitors R-11 and R-12. This is a planned action with a scheduled completion date of October 25, 2003.

– A permanent modification on the containment vacuum and pressure relief control circuits will be installed to restore the design such that the control switch is wired in series with the Train ‘A’ and Train ‘B’ containment isolation relay contacts. This is a planned action with a scheduled completion during the next refueling outage, currently planned for April and May of 2004.

V. ADDITIONAL INFORMATION

A. Failed Component Information:

The failed switch is a GEMCO Switch Model No. 404S-3-2-1-2-2-Y-EE2. The cause of the switch failure has been determined to be age-related degradation and cyclic fatigue.

B. Previous Similar Events:

A review of recent (past 3 years) events at HBRSEP, Unit No. 2, for conditions that could have prevented the fulfillment of a safety function was conducted. There were no events found.