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ACCESSION #: 9906040026

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Edwin I. Hatch Nuclear Plant - Unit 2 PAGE: 1 OF 5

DOCKET NUMBER: 05000366

TITLE: Generator Ground Fault Causes Turbine Trip and Reactor Scram

EVENT DATE: 05/05/1999 LER #: 1999-005-00 REPORT DATE: 05/27/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 98.3

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION: 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Nuclear Safety and
Compliance Manager, Hatch

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: EL COMPONENT: DUCT MANUFACTURER: N/A
REPORTABLE NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 05/05/1999 at 0747 EDT, Unit 2 was in the Run mode at a power level of 2716 CMWT (98.3 percent rated thermal power). At that time, the reactor scrammed and the reactor recirculation pumps tripped automatically on turbine control valve fast closure caused by a turbine trip. The turbine tripped when the main generator tripped on a ground fault. Following the reactor scram, water level decreased due to void collapse from the rapid reduction in power. However, the reactor feedwater pumps maintained water level higher than eight inches above instrument zero. Consequently, no safety system actuations on low level were received nor were any required. Pressure reached a maximum value of 1124 psig; nine of eleven safety/relief valves lifted to reduce reactor pressure. Pressure did not reach the nominal actuation setpoints for the remaining two safety/relief valves. The temperature in the vessel bottom head region decreased by more than the Technical Specification-allowed 100 degrees F in one hour before a recirculation pump could be restarted.

This event was caused by a manufacturer error. Some of the turning vanes located in the discharge duct for the "B" isophase bus duct cooling fan broke loose, shorting a generator phase to ground. The manufacturer installed turning vanes that were not the proper thickness for this application thus resulting in some of their connection points failing. Pieces of the broken vanes were retrieved from the isophase bus duct and the remaining turning vanes were removed from the isophase bus duct cooling system.

END OF ABSTRACT

DISCLAIMER FOR SCANNED DOCUMENTS

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee LLC
Docket No. 50-271 Official Exhibit No. Entergy 9
OFFERED by: Applicant/Licensee Intervenor ______
NRC Staff Other ______
IDENTIFIED on 9/13/06 Witness/Panel Nichols/Casillas
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

Template=SECY-028

SECY-02

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 05/05/1999 at 0747 EDT, Unit 2 was in the Run mode at a power level of 2716 CMWT (98.3 percent rated thermal power). At that time, the reactor automatically scrammed and the reactor recirculation pumps (EIIS Code AD) automatically tripped on turbine control valve (EIIS Code TA) fast closure caused by a main turbine (EIIS Code TA) trip. The main turbine tripped when the main generator (EIIS Code TB) tripped on a ground fault detected simultaneously by generator neutral ground relays (EIIS Code EL) 2S32-R003A, 2S32-R003B, and 2S32-R003C. A recorded ground fault current of 467 amps energized the neutral ground relays; contacts in the energized relays closed causing the generator output breakers (EIIS Code EL) to open. Opening the generator output breakers energized the main turbine trip relays resulting in fast closure of the turbine control valves. Turbine control valve fast closure is a direct input to the reactor protection system (EIIS Code JC) logic system.

Following the automatic reactor scram, vessel water level decreased due to void collapse from the rapid reduction in power. However, the reactor feedwater pumps (EIIS Code SJ) continued to operate limiting the drop in water level. The minimum water level reached during this event was 8.9 inches above instrument zero (167.34 inches above the top of the active fuel), a decrease of approximately 28 inches from a normal level of 37 inches above instrument zero. Vessel water level did not decrease to the actuation setpoint of three inches above instrument zero. Thus, no safety system, including emergency core cooling system, actuations on low (Level 3) water level were received nor were any required.

Vessel pressure reached a maximum value of 1124 psig three seconds after receipt of the scram. Nine of the eleven safety/relief valves actuated to reduce reactor pressure. Vessel pressure did not reach the nominal actuation setpoint of 1140 psig for safety/relief valves 2B21-F013E and 2B21-F013H; therefore, they did not actuate nor were they required to actuate. (Although safety/relief valve 2B21-F013L has a nominal setpoint of 1140 psig, it actuated during this event. The maximum vessel pressure of 1124 psig was within its Technical Specification-allowed setpoint tolerance of 1115.5 psig to 1184.5 psig. Therefore, the safety/relief valve functioned properly during the event.) Vessel pressure was below its pre-event value of 1033 psig within six seconds of the receipt of the scram. All but the four low-low set safety/relief valves closed within nine seconds of the scram; the low-low set safety/relief valves closed as vessel pressure decreased to their nominal closure setpoints of 890 psig, 881 psig, 866 psig, and 851 psig, respectively.

The temperature in the vessel bottom head region, as measured by the vessel bottom head drain line temperature, decreased by 107 degrees F in less than 22 minutes. Unit 2 Technical Specification Limiting Condition for Operation 3.4.9 limits the reactor coolant system cooldown rate to a maximum of 100 degrees F in one hour. At 0810 EDT, Operations personnel restarted one of the reactor recirculation pumps thereby

increasing the bottom head temperature and reducing the bottom head region temperature drop to less than 100 degrees F.

CAUSE OF EVENT

This event was caused by a manufacturer error. Some of the turning vanes located in the discharge duct for isophase bus duct (EIIS Code EL) cooling fan 2R13-C008B broke loose. One or more of the loose pieces shorted a generator phase to the wall of the isophase bus duct, which is grounded. The manufacturer installed turning vanes that were not the proper thickness (gage) for this application thus resulting in some of the vanes failing at their connection points.

The licensed power level and generator output of Unit 2 were increased during the Fall 1998 refueling outage. Larger fans and their associated duct work were installed in the isophase bus duct cooling system during the outage to remove the increased amount of heat generated in the isophase bus resulting from the increased generator output. The discharge ductwork for cooling fan 2R13-C008B included a 90-degree elbow; the elbow was necessary to connect the "B" fan discharge duct to the common header in the isophase bus duct cooling system. (Due to the location of the "A" cooling fan, no elbow was necessary to connect its discharge duct to the cooling system header.) In order to reduce backpressure resulting from the air hitting the side of the 90-degree elbow opposite the fan discharge, and therefore increase the cooling air flow rate, the ductwork manufacturer installed turning vanes in the elbow. This is a standard practice in designing and constructing ductwork. However, the sheet metal used to construct the vanes and the rails used to connect the vanes to the sides of the elbow was too thin for this application.

Twenty-two gage (0.0336") turning vanes were mounted on 24 gage (0.0276") vane rails and tack welded to the rails at two points on two sides. However, it is difficult to weld sheet metal thinner than 18 gauge. Indeed, a visual check revealed that the vanes broke off near the weld points likely due to metal "burn-out" resulting from welding the thin sheet metal. Additionally, portions of the rail also broke loose from the side of the duct at or near the weld points. Visual examination revealed these points likewise had experienced metal burn-out. Although the gage thickness of the turning vanes was in agreement with the Duct Contraction Standard of the Sheet Metal and Air-Conditioning Contractor National Association, the manufacturer should have used thicker sheet metal since welding was used to secure the vanes and rails. Moreover, the required duct specific pressure rating of 17.1 inches water (air velocity of 4400 fpm) should have indicated a thicker sheet metal had to be used to manufacturer the turning vanes and rails. Therefore, the manufacturer erred in using thinner than 18 gage sheet metal for the turning vanes and rails.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. The reactor protection system, an Engineered Safety Feature system, actuated on turbine control valve fast closure when the main turbine tripped following a trip of the main generator from a ground fault. Both reactor recirculation pumps tripped also on turbine control valve fast closure. Nine of eleven

safety/relief valves opened on high vessel pressure; four of the valves continued to operate in the low-low set mode until pressure decreased to their respective closure setpoints.

Fast closure of the turbine control valves is initiated whenever the main generator trips. The turbine control valves close as rapidly as possible to prevent overspeed of the turbine-generator rotor. Valve closing causes a sudden reduction in steam flow that, in turn, results in a reactor vessel pressure increase. If the pressure increases to the pressure relief setpoints, some or all of the safety/relief valves will briefly discharge steam to the suppression pool (EIIS Code BL).

Reactor scram and recirculation pump trip initiation by turbine control valve fast closure prevent the core from exceeding thermal hydraulic safety limits following a main generator or main turbine trip. Closure of the turbine control valves results in the loss of the normal heat sink (main condenser) thereby producing reactor pressure, neutron flux, and heat flux transients that must be limited. A reactor scram is initiated on turbine control valve fast closure in anticipation of these transients. The scram, along with the reactor recirculation pump trip system, ensures that the minimum critical power ratio safety limit is not exceeded.

The recirculation pump trip system, upon sensing a turbine control valve fast closure, trips the reactor recirculation pumps, resulting in a decrease in core flow. The rapid core flow reduction increases void content and reduces reactivity in conjunction with the reactor scram to reduce the severity of the transients caused by the turbine trip.

In this event, the main generator tripped from a ground fault in the isophase bus duct. The main turbine tripped as designed in response to the generator trip. The turbine trip actuated the reactor protection system and scrammed the reactor. All systems functioned as expected and per their design given the water level and pressure transients caused by the turbine trip and reactor scram. Vessel water level was maintained well above the top of the active fuel throughout the transient and indeed never decreased to the Level 3 actuation setpoint. Because the water level decrease was mild, no safety system, including emergency core cooling system, actuations on low water level were received nor were any required.

Typically, the bottom head region of the pressure vessel experiences rapid cooling following a scram coincident with a trip of the reactor recirculation pumps. This cooling is the result of the loss of effective water mixing due to the trip of the recirculation pumps and increased cold water flow from the control rod drive (EIIS Code AA) system following a scram. In this event, the temperature in the vessel bottom head region decreased by 107 degrees F in one hour. However, a bounding analysis indicated cooldown up to 165 degrees F in one hour will not place unacceptable stress on components of the reactor coolant system.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety. The analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Pieces of the broken vanes and rails were retrieved from the isophase bus duct.

The remaining turning vanes were removed from the 90-degree elbow in the "B" cooling fan discharge duct. An evaluation by Southern Company Services ensured that the bus cooling flow requirements remain adequate without the turning vanes. The evaluation also ensured no deleterious effects result with respect to the structural integrity of the ductwork and the increased duty on the fan. The "A" cooling fan discharge ductwork does not contain any turning vanes; therefore, no further modification to its ductwork was necessary or performed.

The licensed power level of Unit 1 was increased during the Spring 1999 refueling outage. However, its existing isophase bus duct cooling system was determined previously to be adequate to handle the increased heat load. Therefore, no modifications were performed on this system during the outage and thus no similar problems are expected and no additional work on the system is required.

Personnel assessed the effects of the excessive cooldown rate on the reactor coolant system as required by Unit 2 Technical Specifications Limiting Condition for Operation 3.4.9, Required Action A.2. An evaluation performed by General Electric in May 1994 (NEDC-32319P) was used in assessing the effects of this event. The May 1994 evaluation, intended to eliminate the need to perform an evaluation for each specific event, demonstrated that reactor pressure vessel and recirculation piping heatup and cooldown rates up to 165 degrees F per hour were acceptable provided certain bounding conditions were met. General Electric and Southern Nuclear personnel reviewed the May 1994 evaluation and concluded that the cooldown of 107 degrees F in one hour experienced during this event was bounded by the generic evaluation. Therefore, personnel determined that the Unit 2 reactor coolant system was acceptable for continued operation.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

This LER does not contain any permanent licensing commitments.

Failed Component Information:

Master Parts List Number: 2R13	EIIS System Code: EL
Manufacturer: Ernest D. Menold, Inc	Reportable to EPIX: Yes
Model Number: N/A	Root Cause Code: B
Type: Turning Vanes	EIIS Component Code: DUCT
Manufacturer Code: None	

There have been no previous similar events in the last two years in which the reactor scrammed while critical.

ATTACHMENT TO 9906040026

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May 27, 1999

Docket No. 50-366 HL-5792

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Generator Ground Fault Causes Turbine Trip and Reactor Scram

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a generator ground fault which caused a turbine trip followed by a reactor scram.

Respectfully submitted,

H.L. Sumner, Jr.

OCV/eb

Enclosure: LER 50-366/1999-005

cc: Southern Nuclear Operating Company
Mr. P.H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L.N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L.A. Reyes, Regional Administrator
Mr. J.T. Munday, Senior Resident Inspector - Hatch