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June 2, 1992

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report

This letter forwards one (1) copy of Licensee Event Report 92-005.

Sincerely,

John J. Barton
Vice President and Director
Oyster Creek

JJB/BDEM:jc
Enclosure

cc: Administrator, Region 1
Senior NRC Resident Inspector
Oyster Creek NRC Project Manager

(LER-COVLTRS)

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Reactor Scram & Engineered Safety Features Actuations Caused by Offsite Fire

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
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0 5	0 3	9 2	9 2	— 0 0 5 —	0 0	0 6	0 2	9 2		0 5 0 0 0

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

OPERATING MODE (9):
POWER LEVEL (10): 1 0 0

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 below and in Text, NRC Form 366A)
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)	X 50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Lynne Munzing	6 0 9	9 7 1 - 4 3 8 1

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)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

A reactor scram and subsequent Engineered Safety Features systems actuations were caused by a turbine load rejection due to faults on off-site 230kV transmission lines caused by a forest fire. The scram occurred at 1326 hours on May 3, 1992 and the event concluded at 0635 hours on May 4, 1992. The reactor was operating at approximately 100% power before the scram. Numerous other engineered safety features actuated including Isolation Condensers, Containment Isolation, Diesel Generator fast start, Core Spray and Standby Gas Treatment. Several additional scram signals occurred in the process of bringing the plant to cold shutdown and returning power supplies to off-site sources. An Unusual Event was declared based on high drywell temperature, and an Alert was declared based on the potential of the forest fire to further affect the plant. The plant was brought to cold shutdown at 2234 hours on May 3, and the emergency condition was terminated at 0635 hours on May 4, after off-site power was restored to vital electrical buses. Off-site power had been available since 1331 hours on May 3, but plant management decided not to place the vital buses on off-site power until reliability could be assured. No plant structures or equipment were damaged by the fire. The forest fire which caused the loss of off-site power was the root cause of the event, and the safety significance was minimal because all systems functioned as required. Corrective actions include a revision to the Diesel Generator operating procedure to prevent an avoidable scram when securing diesel generator operation. Utility personnel inspected off-site power lines and found no damage. High resistance contacts on the control rod drive pump time delay relay were replaced due to the pump's failure to start on a diesel generator load sequence.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

DATE OF OCCURRENCE

The event began on May 3, 1992, at 1326 hours and concluded on May 4, 1992, at 0635 hours.

IDENTIFICATION OF OCCURRENCE

A reactor scram and subsequent Engineered Safety Features systems actuations were caused by a turbine load rejection due to faults in off-site 230kV transmission lines. This is reportable in accordance with 10 CFR 50.73 (a)(2)(iii) and (a)(2)(iv).

CONDITIONS PRIOR TO OCCURRENCE

The reactor was critical in the RUN mode at 1920 megawatts thermal (99.5% full power). Xenon buildup was in progress following recovery from a power reduction for Main Steam Isolation Valve (IEEE-SB, CFI-ISV) testing. The turbine-generator (IEEE-TA, CFI-TRB) was on line at 641 megawatts electric with automatic voltage control. Reactor recirculation (IEEE-AD) flow was 15E4 gpm with five pumps in service. Reactor pressure was 1020 psig and level was 160" TAF (above top of active fuel). Primary containment was intact and inerted.

DESCRIPTION OF OCCURRENCE

At 1310 hours on May 3, 1992, a maintenance supervisor reported to the Control Room that a forest fire was burning west of the plant. Security and Operations Department personnel were assigned to observe the fire and the system dispatcher was notified due to the close proximity of the fire to the 230kV distribution lines. At 1325 hours electrical fluctuations were observed and 4160 volt vital electric bus (IEEE-EB) low voltage alarms were received on the Plant Computer System (IEEE-ID), but not on the Control Room annunciators.

At 1326:30, a full reactor scram occurred, caused by operation of the turbine controls acceleration relay (IEEE-JJ, CFI-RLY). The Turbine controls acceleration relay operation resulted from a rapid load rejection which occurred after off-site distribution breakers (CFI-52) tripped due to faults apparently from heavy smoke and heat in the vicinity of the off-site 230kV line insulators. It is believed that these smoke and heat conditions resulted in ionization of the air around the insulators (CFI-INS), causing arcs. The 34.5 kV lines (IEEE-EA) which supply Startup Transformers S1A and S1B (CFI-XFMR) were also lost, resulting in a complete loss of off-site power. When the generator tripped, generator output breakers GC1 and GD1 (IEEE-EL) tripped open, 4160V main breakers 1A and 1B (IEEE-EA) (non-safety-related buses) tripped open, and Startup Transformer breakers S1A and S1B closed to supply the plant with off-site power, although there was no off-site power available (see attached Electrical Distribution schematic diagram). The diesel generators (IEEE-EK, CFI-DG) , which had already received a signal to start and idle on the generator trip, received fast start signals at 1326:34 from low-low voltage signals on safety-related 4160V buses 1C and 1D (IEEE-EB).

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The diesel generators and the loads sequenced as designed, except for Control Rod Drive Pump A.

After the reactor scram, reactor high pressure (a scram signal) and Reactor Recirculation pump trips occurred. Electromatic Relief Valves (CFI-RV) (EMRVs) A and D opened on high pressure (1060 psig). A reactor low level scram signal was then received due to rapid void collapse. Isolation Condensers (IEEE-BL) actuated at 1326:33 from the reactor high pressure signal. The reactor high pressure signal cleared at 1326:36, and EMRVs A and D closed. The reactor low level signal cleared at 1326:46. The Standby Gas Treatment System (SGTS) (IEEE-BH) initiated at 1326:46, apparently due to spurious radiation alarms resulting from voltage transients as the Diesel Generators restored vital bus power. The low-low voltage alarms on safety-related 4160V buses 1C and 1D cleared at 1326:51. Two reactor low level alarms were received and level was approaching the low-low level setpoint, so the Main Steam Isolation Valves (MSIVs) were manually closed at 1328:57 in anticipation of a reactor isolation signal. The reactor low-low level signal was then received at 1329:44 and initiated both Core Spray Systems (IEEE-BM). Water was not injected into the reactor vessel due to the pressure interlock. A pressure increase due to removal of Isolation Condensers from service to control reactor pressure caused a void collapse which resulted in the low-low reactor water level condition. As the Isolation Condensers were cycled in and out of service for reactor pressure control, numerous reactor high and low level alarms and scram signals were received. The Alternate Rod Injection System (ARI) (IEEE-AA) initiated on reactor low-low level at 1333:51.

Off-site power became available to the Startup Transformers at 1331:03. At 1332, 4160V buses 1A and 1B were re-energized from the Startup Transformers. Upon power restoration to these non-safety-related 4160V buses, Circulating Water Pumps (IEEE-KE), Condensate Pumps (IEEE-SD), Feedwater Pumps (IEEE-SJ) and Air Compressors (IEEE-LD) were restarted. A decision was made by plant management not to place the safety-related 4160V buses on off-site power until reliability could be assured. Fires continued to burn near the 230 kV lines.

As required by Emergency Operating Procedures (EOPs), the Feedwater Pumps were started. Their feed regulating valves (CFI-FCV) were locked up in the open position due to the loss of air. Air compressors tripped on loss of offsite power and do not automatically load on a diesel start sequence. Due to the significant number of continuous alarms, the entry into EOPs and restoration of off-site power, the operator did not recognize that the valves were locked up and failed to close in response to a manual closure signal. This caused a high reactor water level, requiring the Isolation Condensers to be removed from service to prevent water hammer. EMRVs A and B were opened to control reactor pressure and reduce reactor level. The Containment Spray System (IEEE-BO) was started in the torus cooling mode due to the discharging EMRVs.

The associated Emergency Service Water (IEEE-BS) pump started 45 seconds after the Containment Spray Pump, as designed. Both EMRVs were soon closed and the high reactor water level condition cleared.

At 1402 hours the Group Shift Supervisor in the Control Room declared an Unusual Event based on indicated high drywell temperature of 160°F. The scram and ARI were reset. The Group Shift Supervisor then declared an Alert at 1434 due to the potential for the off-site fire to further affect the plant. The Emergency Response organization was activated.

At 1455 the reactor isolation signal was reset. Several low level scram signals in succession were received while maintaining reactor level in the desired band. At 1609 the Containment Spray System was taken out of the torus cooling mode and returned to standby readiness. Isolation Condenser logic was reset at 1742, and Shutdown Cooling (IEEE-BO) was placed in service at 1945. The Main Steam Isolation Valves were opened at 2044 to vent the reactor. The reactor reached cold shutdown conditions at 2234.

At 0240 on May 4, a reactor scram and containment isolation signal were received when power was lost to 4160V bus 1D while securing Diesel Generator 2.

At 0505 the emergency classification was downgraded to an Unusual Event. By 0631 both 4160V buses 1C and 1D were restored to their normal off-site power supplies and the associated Diesel Generators shutdown. The plant secured from the Unusual Event at 0635 hours on May 4, 1992.

No plant structures or equipment were directly affected by the fire. The fire did approach within approximately 70 feet of the Fire Pump House (IEEE-KP), which is located southwest of the main plant site and across the salt water discharge canal. Local fire department and plant personnel were stationed at the Fire Pump House during the period that it was threatened.

ANALYSIS OF OCCURRENCE AND SAFETY SIGNIFICANCE

The generator load rejection scram anticipates the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves (CFI-FCV) due to a load rejection. The scram functioned appropriately on a load rejection signal and all control rods fully inserted.

The Diesel Generators are designed to start and automatically load all safety related pumps and auxiliaries required for safe shutdown of the reactor in the event of a design basis accident with a loss of off-site power. All required loads started automatically except Control Rod Drive (CRD) Pump A (IEEE-AA). The significance of this failure to start is minimal, since the other CRD Pump did start.

The high pressure and low-low reactor water level after the scram initiated the Isolation Condensers and EMRVs as designed. The Isolation Condensers remove core residual and decay heat, and depressurize the reactor vessel in the event the main condenser is not available as a heat sink. Both Isolation Condensers initiated and functioned as designed. The EMRVs provide overpressure protection to avoid unnecessary safety valve actuation during plant transients that result in a pressure increase. EMRVs A and D opened appropriately when their setpoint of 1060 psig was reached.

Restart of Reactor Feedwater pumps with their regulating valves locked open caused a high reactor water level, requiring removal of the Isolation Condensers from service. EMRVs were successfully used to control reactor pressure until level returned to the desired control band.

Due to heavy concentration of smoke in the area an assessment of equipment that might be affected by the smoke was warranted. Engineering analysis determined that operation in a smoke environment did not adversely affect the Diesel Generators or Diesel Fire Pumps. A sample charcoal canister from the Standby Gas Treatment System was removed and sent for laboratory analysis. The results indicated no damage to the charcoal beds from the fire's smoke.

All other automatic functions actuated and operated as designed, therefore, safety significance of this event is considered minimal.

APPARENT CAUSE OF OCCURRENCE

The cause of the load rejection scram was the loss of off-site power initiated by a forest fire. When off-site power was lost, the turbine controls acceleration relay responded to rapidly close the control valves to prevent a turbine over speed condition. The rapid response by the acceleration relay was sensed by the Reactor Protection System, which in turn produced a scram.

The cause of the scram and isolation signal at 0240 hours on May 4 was an inadequate procedure. A surveillance procedure contained appropriate instructions to prevent a reactor scram when securing diesel generators, but the operating procedure did not contain the same instructions. In addition, due to inadequate self-checking, the operator was monitoring the incorrect voltage indicator while securing Diesel Generator 2; the voltage indicators labeled "DG" and LINE" are actually reversed during this electrical configuration.

The cause of the failure of Control Rod Drive Pump A to start on the Diesel Generator loading sequence was a set of high resistance contacts on the time delay relay (CFI-2) for pump start on the automatic loading sequence.

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CORRECTIVE ACTION

Utility personnel inspected off-site power lines prior to placing the generator on line and found no damage. The diesel generator operating procedure will be revised to include steps to prevent a scram signal when securing diesel generator operation, and the revised version of the procedure is currently being reviewed with operators on the non-certified plant referenced simulator (operators are participating in simulator development). The high resistance contacts on the Control Rod Drive pump time delay relay were replaced.

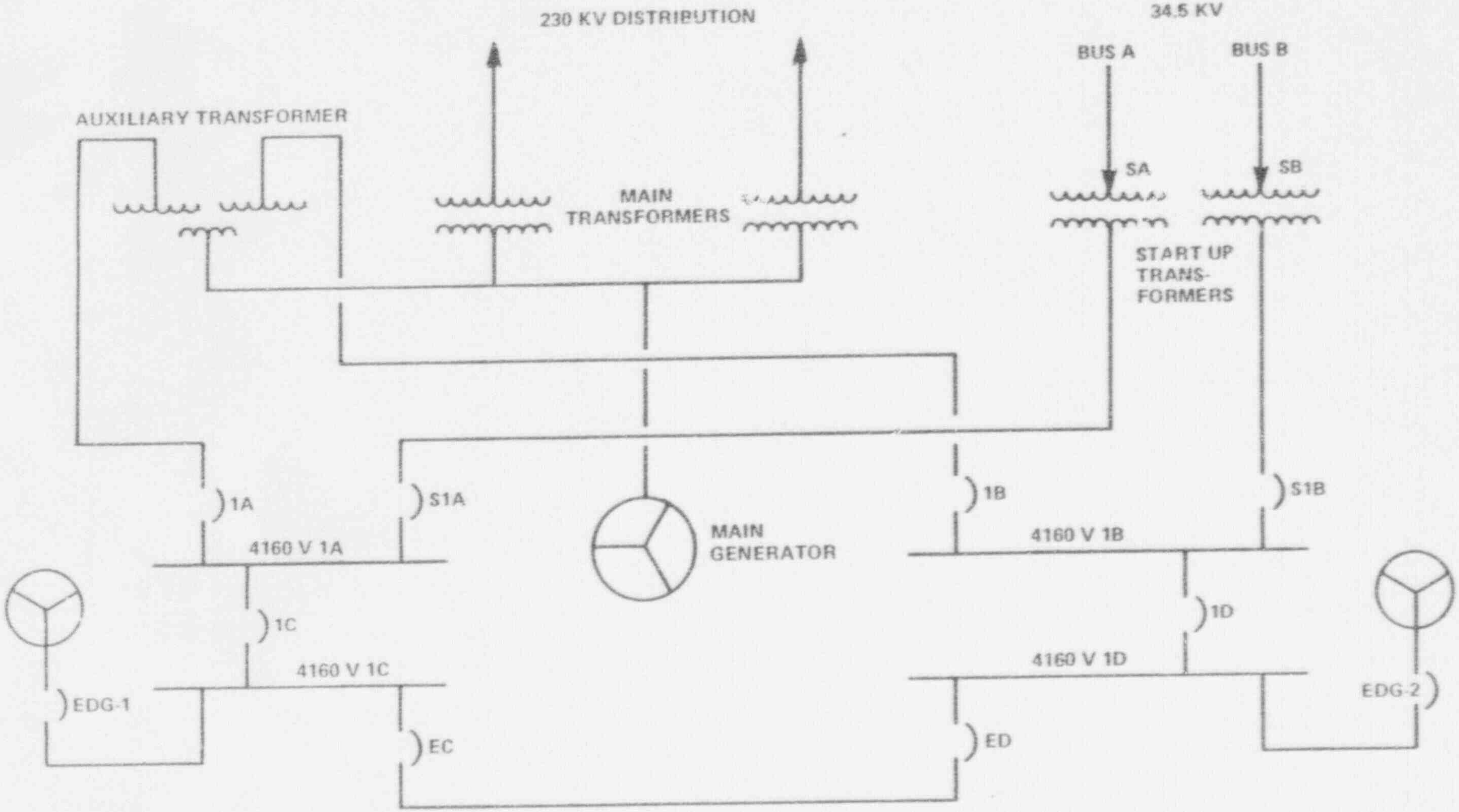
SIMILAR EVENTS

LER 91-005 Automatic Reactor Scram Due to Loss of Feedwater Flow Caused by a Grounded Condensate Pump Motor

LER 89-016 Main Transformer Failure Causes Automatic Reactor Shutdown

LER 89-015 Main Generator Trip Causes Automatic Reactor Shutdown Due to Personnel Error

LER 87-11 High RPV Level Trip/Scram Caused by Lost Feedwater Flow Signal Due to Procedural Inadequacy and MSIV Auto Closure Due to Loose Wire


230 KV DISTRIBUTION
34.5 KV
BUS A
BUS B
AUXILIARY TRANSFORMER
MAIN TRANSFORMERS
SA
SB
START UP TRANS-FORMERS
1A
S1A
1B
S1B
MAIN GENERATOR
4160 V 1A
4160 V 1B
1C
1D
4160 V 1C
4160 V 1D
EDG-1
EDG-2
EC
ED