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FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269

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SUBJECT: LER 89-002-00:on 890103,fire in 1TA switchgear due to unknown cause.

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(9-83)

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APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Oconee Nuclear Station, Unit 1 | 0 5 0 0 0 2 6 9 | 1 OF 1 9 |

TITLE (4)
Fire in 1TA Switchgear Due to Unknown Cause

| EVENT DATE (5) MONTH | DAY | YEAR | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | REPORT DATE (7) MONTH | DAY | YEAR | OTHER FACILITIES INVOLVED (8) FACILITY NAMES | DOCKET NUMBER(S) |
|---|---|---|---|---|---|---|---|---|---|---|
| 0 1 | 0 3 | 8 9 | 8 9 | — 0 0 2 — | 0 0 | 0 2 | 0 2 | 8 9 | | 0 5 0 0 0 |
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POWER LEVEL (10): 2 6

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| 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) | |
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EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

On January 3, 1989 with Unit 1 at 26% power during power escalation after a previous trip, a fire occurred in 1TA switchgear. The switchgear fire resulted in the tripping of two Reactor Coolant Pumps (RCP) fed from the 1TA switchgear. An Unusual Event was declared. The Reactor was manually tripped and the two remaining RCPs were secured in preparation for using water to extinguish the switchgear fire. Cooldown rates specified by Technical Specifications were violated as a result of overfeeding the SGs while mitigating a RCS pressure increase. After the fire was distinguished, the RCS was not operated within the Thermal Shock Operating Region(TSOR). The root cause of the switchgear fire was Equipment Failure of unknown cause. The root cause of the vessel overcooling was Personnel Error. The root cause of the RCS not being operated in the TSOR region for the required time period is Management Deficiency. The root cause of the TS violation for exceeding the temperature differential between the pressurizer water and the spray fluid was Management Deficiency. The subsequent corrective actions were to provide additional guidance to the operators on entering the TSOR region and when it was necessary to do so. In addition, an analysis of the pressurizer spray nozzles steam generator and the reactor vessel was performed to evaluate the impact of the temperature transients throughout the event, adequate core subcooling margin was maintained and fuel integrity was not threatened. In addition, the primary system integrity was not breached and a release of radioactivity did not occur. The health and safety of the public were not affected.

IE22





☆ U.S.GPO:1984-0-454-481/18759

INTRODUCTION

On January 3, 1989, with Units 2 and 3 at 100% power and with Unit 1 at 26% during power escalation after a previous trip, a fire occurred in 1TA switchgear. The switchgear fire resulted in the tripping of two Reactor Coolant Pumps (RCP) fed from the 1TA switchgear. After unsuccessfully fighting the fire for approximately thirty (30) minutes, the Shift Supervisor activated the Technical Support Center (TSC) and the Operational Support Center (OSC), and declared an Unusual Event. The Reactor was manually tripped and the two remaining RCPs were secured in preparation for using water to extinguish the switchgear fire after carbon dioxide and dry chemicals had proven unsuccessful. After the RCPs had been tripped, the Integrated Control System (ICS) failed to control Steam Generator (SG) level at the proper setpoint using the Auxiliary Feedwater nozzles. Control Room Operators took manual action to correct the ICS failure and control Reactor Coolant System (RCS) pressure which peaked at 2395 psig. Cooldown rates specified by Technical Specifications were violated as a result of overfeeding the SGs while mitigating the RCS pressure increase.

After the fire was extinguished the TSC decided that RCS pressure should not be reduced so that RCS parameters were within the limits of the Thermal Shock Operating Region (TSOR), the Unusual Event was terminated, a RCP started, a three hour RCS soak was completed, and a plant cooldown commenced. In an effort to reduce RCS pressure during the event, use of Auxiliary Pressurizer Spray (APS) was attempted. However, due to a closed valve in containment, use of APS was not possible. Later initiation of APS led to a Technical Specification (TS) violation.

The root cause of the 1TA switchgear fire was Equipment Failure of unknown cause. ITE-Gould manufacturing representatives and Design personnel were involved in the analysis of the switchgear and could not determine a failure mode although it was possible to postulate some theories.

The root cause of the vessel overcooling was Personnel Error, Improper Action. This is based on the fact that the Control Room Operators were monitoring an increase in RCS pressure and Incore Thermocouples and did not take action to monitor cold leg temperature indications and Technical Specification cooldown rates. There were contributing causes of ineffective training and operator burden.

The root cause of the RCS not being operated in the TSOR region for the required time period is Management Deficiency. This resulted from the fact that the Emergency Operating Procedure (EOP) contains criteria which should be used to determine if a three hour soak in the TSOR region is necessary. The Technical Support Center made the decision that operation in the TSOR region was not required based on an incorrect interpretation of the EOP

guidance.

The root cause of the TS violation for exceeding the temperature differential between the pressurizer water and the spray fluid was Management Deficiency, Deficient Supervision. This is due to an Operations Coordinator instructing the Control Room Supervisor to initiate the APS during troubleshooting.

The subsequent corrective actions were to provide additional guidance to the operators on entering the TSOR region and when it was necessary to do so. Babcock and Wilcox performed an analysis of the Pressurizer Spray Nozzle to evaluate the impact of the temperature transient. This analysis concluded that no significant degradation had occurred.

SEQUENCE OF EVENTS

| Date/time | | Description |
|---|---|---|
| January 3, 1989 | | |
| 19:16 | _ | Reactor power was at 26%. |
| 19:16 | _ | 6900 V auxiliaries manually transferred from the Startup Transformer to the Main Transformer (1T). |
| 19:16:36:780 | _ | 1T Differential alarm on the Z phase received. |
| 19:16:36:789 | _ | 1T Differential alarm on the X phase received. |
| 19:16:36:790 | _ | Generator Lockout occurred. |
| 19:16:36:821 | _ | Main Turbine tripped. |
| 19:16:37:351 | _ | 1A1 Reactor Coolant Pump (RCP) tripped. |
| 19:16:37:417 | _ | 1B1 RCP tripped. |
| 19:17 | _ | Fire alarm statalarm received. |
| | _ | Received telephone reports of fire and an explosion at 6900 V switchgear. |
| | _ | Reactor ran back to 14% power. |
| | _ | Fire Brigade was dispatched to 6900 V switchgear. |
| | _ | Verified that 1TA isolated from 6900 V sources. |
| 19:29 | _ | 1TA DC control power removed at 1DIA and 1DIB. |
| 19:33 | _ | Use of carbon dioxide extinguishers failed to extinguish 1TA fire. |

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

| Date/Time | | Description |
|---|---|---|
| 19:41 | - | Use of dry chemical extinguisher also failed to extinguish 1TA fire. |
| 19:45 | - | Shift Supervisor (SS) declared an Unusual Event. |
| 19:55 | - | Started Reactor power reduction. |
| 19:56 | - | 1A High Pressure Injection Pump (HPI) manually started. |
| 19:57 | - | SS activated the Technical Support Center (TSC) and the Operational Support Center (OSC). |
| | - | SS requested additional Fire Brigade support from off duty shift personnel. |
| 19:58 | - | HPI Pump suction from the BWST valve (1HP-24) opened. |
| | - | Reactor Coolant loop A injection valve (1HP-26) throttled open. |
| 19:59 | - | Fire Brigade leaders and SS decided to use water fog on fire. |
| 19:59:24 | - | Reactor manually tripped from 4% power. |
| 20:02:18 | - | 1RC-1 (Pressurizer Spray Valve) opened at 2205 psig setpoint. |
| 20:02 | - | 1A2 RCP manually tripped. |
| | - | 1B2 RCP manually tripped. |
| | - | 1TB deenergized to allow water to be used on fire. |
| | - | Integrated Control System (ICS) failed to raise Steam Generator (SG) levels to 50% on operating range. |
| | - | ICS failed to swap Feedwater (FDW) to the Auxiliary FDW nozzles. |
| 20:04 | - | Reactor Protective System high pressure trip setpoint (2355 psig) reached. |
| | - | While in "MANUAL" Turbine Bypass Valves (TBV) throttled to 10% open. |
| 20:05 | - | Control Room Operators "A" and "B" (CRO) took manual action to increase SG levels to 50%. |
| | - | CROs took manual action to swap FDW to the Auxiliary FDW nozzles but left Main FDW startup block valves open. |
| | - | TBV placed in AUTO thereby closing valves. |
| 20:06:02 | - | 1HP-26 throttled full open. |

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| Oconee Nuclear Station, Unit 1 | 0 5 0 0 0 2 6 9 | 8 9 | — 0 0 2 | — 0 0 | 0 5 OF 1 9 |

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

| Date/Time | | Description |
|---|---|---|
| January 3, 1989 (continued) | | |
| 20:06:16 | – | 1HP-26 closed. |
| 20:06:57 | – | 1A HPI pump was secured. |
| 20:10 | – | Operations Shift Supervisor "A" and Shift Engineer "A" determined that Thermal Shock Operating Region (TSOR) requirements had been met. |
| | – | CRO attempted to use HPI Auxiliary Pressurizer Spray to depressurize but could not establish flow. |
| | – | 1A HPI pump restarted automatically. |
| 20:15 | – | Fire was reported to be out. |
| 20:20 | – | Closed "1A" SG Normal Startup Header Block valve (1FDW-36) and "1B" SG Normal Startup Header Block valve (1FDW-45) to isolate FDW to Main FDW valves. |
| 20:22 | – | TBV throttled to 15% open. |
| | – | 1HP-24 was closed. |
| 20:29 | – | Main FDW Control Valves placed in MANUAL. |
| 20:31:11 | – | CRO pressed closed pushbutton for "1A" SG Main Block valve (1FDW-31). |
| 20:31:18 | – | CRO pressed closed pushbutton for "1B" SG Main Block valve (1FDW-40). |
| 20:33:05 | – | 1FDW-40 reached the CLOSED position. |
| 20:33:12 | – | 1FDW-40 started going OPEN. |
| 20:34:57 | – | 1FDW-40 reached full OPEN. |
| 20:35:12 | – | CRO pressed CLOSED pushbutton for 1FDW-40. |
| 20:35:52 | – | Main FDW pumps tripped on high SG level. |
| 20:35:58 | – | After reaching the "CLOSED" position 1FDW-40 began to go OPEN. |
| 20:36 | – | All Emergency FDW pumps started automatically. |
| 20:41:16 | – | 1FDW-40 reached the OPEN position. |
| 20:41:35 | – | CRO took action to close 1FDW-40. |
| 20:43:28 | – | After reaching the "CLOSED" position 1FDW-40 breaker was opened. |
| 21:03 | – | 1TB switchgear was reenergized from CT-1. |
| 21:19 | – | TSC was established. |
| | – | Station Manager relieved the SS as the Emergency Coordinator (EC). |
| | – | TSC started discussion about whether operation within the Thermal Shock Operating Region (TSOR) was required. |
| 21:50 | – | TSC decided that operation in the TSOR was not required. |

| Date/Time | Source | Description |
|---|---|---|
| January 3, 1989 (continued) | | |
| 21:50 | _ | TSC decided that a three hour soak would be performed at stable RC pressure and temperature after RCP restart. |
| 22:00 | _ | Operations could not establish seal return flow to 1A2 RCP. |
| 22:03 | _ | 1B2 RCP was restarted. |
| | _ | Three hour soak was started. |
| January 4 | | |
| 01:05 | _ | Three hour soak was completed. |
| 01:18 | _ | EC terminated Unusual Event. |
| 02:02 | _ | 1A Condensate Booster Pump was started to feed the SG. |
| | _ | Emergency FDW pumps were secured. |
| 02:45 | _ | Entry into Reactor Building found HPI to Auxiliary Pressurizer Spray line valve (1HP-340) shut. |
| | _ | 1HP-340 was reopened. |
| | _ | Operations Coordinator "A" directed shift personnel to initiate Auxiliary Pressurizer Spray (APS) to verify flow. |
| 03:25 | _ | Seal return flow was established on 1A2 RCP. |
| | _ | 1A2 RCP was restarted. |
| | _ | Normal plant shutdown commenced. |

BACKGROUND

An event where the Reactor Vessel [EIIS:RCT] is subjected to an overcooling transient followed by a repressurization or simultaneous pressurization of the system is defined as Pressurized Thermal Shock (PTS). In the event of a PTS concern, a Thermal Shock Operating Region (TSOR) has been defined. This is a region bounded by pressure and temperature ranges which should be entered to minimize any possible degradation which the PTS transient could cause to the vessel integrity. Entering this region involves RCS [EIIS:AB] depressurization to within specified bounds. The guidance given by EP/1/A/1800/01 (Emergency Operating Procedure) states that the Reactor Coolant System (RCS) pressure will be maintained in the TSOR when:

a. RCS temperature less than 500 Degrees-F and all Reactor Coolant Pumps (RCP) are off and High Pressure Injection (HPI) [EIIS:BQ] has operated in the Engineered Safeguards [EIIS:JE] mode or;
b. The cooldown rate exceeds or has exceeded 100 Degrees-F per hour and a 100 Degrees-F temperature change has occurred.

EOP guidance states that the reactor will be maintained in the TSOR region for a three hour soak whenever entry into TSOR is required. This time

period allows the temperature of the vessel wall to stabilize, thus relieving stresses across the wall.

The purpose of the 6900 V Auxiliary Power System is to provide power to the RCPs. The Auxiliary Power System is divided into two switchgear [EIIS:SWGR] groups designated as 1TA and 1TB, each of which provide power to one RCP in each of the RCS loops. These power sources are not designated as safety related.

The Pressurizer [EIIS:PZR] is used to control pressure in the RCS by shrinking and expanding the steam bubble. A spray line fed from the 1A1 RCP, for Unit 1, provides the ability to spray down the steam bubble and thereby shrink the bubble and decrease the RCS pressure. If the 1A1 RCP (and subsequently Pressurizer Spray) should become unavailable, and it is desired to decrease RCS pressure, the operator has three alternatives which he may use. The first alternative is to depressurize the RCS via the Power Operated Relief Valve (PORV) which vents the Pressurizer to the Quench Tank. The second alternative is to use Auxiliary Pressurizer Spray which is supplied through the HPI system. The third alternative is through use of the Turbine Bypass valves. The valves may be throttled open to allow the Steam Generators [EIIS:SG] to steam off thereby removing more of the heat content from the RCS and shrinking the RCS inventory.

The Integrated Control System (ICS) [EIIS:JA] has two automatic actions which it will initiate upon loss of all RCPs with Main Feedwater (FDW) [EIIS:SJ] available. The first of these actions is to swap FDW from the Main FDW nozzles to the Auxiliary nozzles. The second of these actions is to raise the level in the Steam Generators (SG) to 50% on the operating range. These actions are taken to raise the thermal center of the SG so that natural circulation of the RCS can be established.

The Technical Support Center (TSC) and the Operational Support Center (OSC) are established in the case of a plant emergency or at the discretion of the Shift Supervisor. The TSC provides assistance to Operations personnel to assist in bringing the plant to safe shutdown conditions. The OSC provides craft support for resolution of problems which are discovered and need prompt resolution to allow the plant to be maintained in a safe condition. The Emergency Coordinator is responsible for overseeing the activities of both the TSC and the OSC and ensuring that the plant is being maintained in a safe condition. Normally the Emergency Coordinator is the Station Manager.

Technical Specification 3.1.2.1 gives the limits for RCS pressure These limits are given in Table 3.1-1 and list the maximum cooldown rate as less than 50 Degrees-F in any thirty minutes when temperature is greater than 280 Degrees-F.

Technical Specification 3.1.2.6 gives the cooldown limits for the

Pressurizer and will not exceed 100 Degrees-F per hour. This specification also lists the temperature differential between the Pressurizer and the spray fluid is greater than 410 Degrees-F.

DESCRIPTION OF EVENT

On January 3, 1989, at approximately 1916 hours with Unit 1 at 26% during power escalation after startup following a trip, a fire occurred in the Unit 1 6900 V switchgear. The fire occurred while swapping the power supply for the 6900 V switchgear from the startup transformer (CT-1) to the main transformer (1T). Once the swap had occurred, Phase Differential alarms were received on 1T and a Generator lockout occurred. This resulted in a Main Turbine trip on a Generator lockout and a Reactor runback to 14% power. Control Room Operators (CRO) "A", "B" and "C" also received telephone reports of a fire and explosion in the vicinity of the 6900 V switchgear. A control room statalarm for the fire detection system was also lit. The Fire Brigade (FB) was dispatched to the 6900 V switchgear and operators verified, from the control room, that 1TA was isolated from the 6900 V sources. At 1929 hours, operators removed power from the DC control circuits to 1TA to complete the electrical isolation of the switchgear.

At 1933 hours, the Fire Brigade attempted to extinguish the fire using carbon dioxide extinguishers but were unsuccessful in the attempt. A second attempt was made using dry chemical extinguishers. This attempt was also unsuccessful due to the high heat content of the cabinet causing reflash. After the failure of both attempts to extinguish the fire and due to the time elapsed since the fire was discovered, the Operations Shift Supervisor (SS) declared an Unusual Event and activated the Technical Support Center (TSC) and the Operational Support Center.

At 1956 hours, 1A High Pressure Injection (HPI) pump was manually started making a total of two HPI pumps operating. This action was taken because reactor power was being reduced by the CROs and the Reactor Coolant System (RCS) temperature was dropping. This temperature decrease was causing a corresponding drop in Pressurizer level due to shrinkage in the RCS. At 1957 hours, the SS requested that off-duty shifts be called in to provide backup to the Fire Brigade members. Due to the lack of success when fighting the fire with carbon dioxide and dry chemicals, the decision was made to fight the fire using a water fog. In preparation for this and to provide personnel safety, the Reactor was manually tripped at approximately 4% power and the two remaining RCPs were tripped in preparation for deenergizing 1TB. The remaining 6900 V switchgear (1TB) was then deenergized. After the remaining two RCPs were tripped, the Integrated Control System (ICS) failed to take automatic action to swap Feedwater

(FDW) to the Steam Generator (SG) Auxiliary nozzles and increase the SG levels to 50% on the operating range. The ICS failure was due to signal cables which had been damaged by the switchgear fire and were sending erroneous signals to the ICS. The CRO manually raised the SG levels and swapped FDW to the Auxiliary SG nozzles but failed to isolate FDW to the Main nozzles. This action resulted in a further drop of RCS temperature. The RCS pressure reached a maximum peak of 2395 psig during the transient.

At this time the CRO secured the 1A HPI pump due to increasing pressure and placed it in "AUTO". The Turbine Bypass Valves (TBV) were also placed in "Manual" and throttled to approximately 10% open to reduce RCS pressure. The valves were left open for approximately one minute then returned to "AUTO" which subsequently closed the valves. An RCS pressure reduction was attempted through use of the HPI Auxiliary Pressurizer Spray (APS); however, flow could not be established. At this time the CRO noted an RCS pressure reduction resulting from the increasing SG levels. The CRO restarted the 1A HPI pump to compensate for the pressure reduction. This created another RCS pressure spike of approximately 2300 psig due to an increase in RCS makeup flow.

At 2015 hours, the fire was reported to be extinguished. TBV were then placed in "MANUAL" and throttled to approximately 15% open to stabilize RCS pressure. RCS pressure stabilized at approximately 2100 psig. During this time , Shift Supervisor "A" and Shift Engineer "A" determined that the requirements for operation in the Thermal Shock Operating Region (TSOR) had been met and began to determine means of regaining HPI APS to depressurize into the TSOR region. To ensure that valve position would not change, CROs then placed the Main FDW Control Valves in "MANUAL". However, when this was done, the "A" Main FDW Control Valve backed off its closed seat and allowed an overfeed of the "A" SG to begin. CROs initiated action to close the block valves for both of the SGs. The "B" SG block valve (1FDW-40) started closed. However, due to a failed yoke bearing on the "A" SG block valve (1FDW-31), it did not close fully and the overfeed of the "A" SG continued. Both FDW pumps tripped on high "A" SG level in less than seven minutes thereby stopping the "A" SG overfeed. 1FDW-31 was later closed from the breaker. At the time of the FDW pump trip, RCS cold leg temperature was approximately 465 Degrees-F 1FDW-40 cycled twice due to a stuck pushbutton on the control board before it was finally maintained closed by opening the breaker. Upon the loss of both FDW pumps, Emergency FDW was automatically initiated. All pumps started and functioned to maintain SG levels.

After the Transmissions Department had completed a checkout of the 1TB switchgear, it was reenergized from CT-1 and no problems were discovered. At 21:19 the TSC was established and the Station Manager relieved the SS as the Emergency Coordinator. A discussion was started in the TSC about the

need to operate the RCS in the Thermal Shock Operating Region (TSOR). Unit Engineer "A" reported that the criteria requiring a three hour soak in the TSOR had been met and that the RCS should be maintained in the TSOR region. However, due to the lack of understanding of the procedural guidance specifying the start point of the temperature decrease and the misunderstanding by the TSC that the ES criteria could only be satisfied with three HPI pumps running, the decision was reached that operation in the TSOR was not required. It was decided that a three hour soak at stable RCS pressure and temperature would be performed after an RCP restart. At 2203 hours, 1B2 RCP was restarted and the operators maintained the reactor at stable RCS conditions for three hours. The soak was completed at 0105 hours on January 4, 1989. The Emergency Coordinator terminated the Unusual Event at 0118 hours.

At approximately 0200 a reactor building entry discovered that valve 1HP-340, HPI to Auxiliary Pressurizer Spray line, was shut and had prevented flow to the Pressurizer Auxiliary Spray. The valve was reopened and Operations Coordinator "A" instructed shift personnel to initiate HPI APS to verify that this had corrected the problem which had prevented the use of APS during the incident. At the time of the initiation, the temperature differential between the pressurizer water and the spray fluid was greater than the Technical Specification limit of 410 Degrees-F.

The 1A Condensate Booster Pump was started to feed the SGs and the Emergency FDW pumps were secured. At 0325 hours the 1A2 RCP was restarted after seal return flow had been established by cycling the AC and DC oil lift pumps, reducing Letdown Storage Tank pressure, and isolating RCP seal return flow on the idle RCPs. A normal plant shutdown then commenced.

Concerning reactivity control, the reactor was tripped manually and all Control Rod Drive breakers opened satisfactorily with all Control Rods fully inserting into the core.Following the trip of the reactor, the RCS pressure was maintained in the range of from 1995 psig to 2395 psig. The maximum RCS temperatures occurred at the start of the reactor power reduction, at which time the average temperature was 575 Degrees-F. The minimum value of the average temperature over the time span for which hot leg temperature data are available is 490 Degrees-F. However the RCS cold leg temperature decreased to 426 Degrees-F at 59 minutes after the reactor trip and fell to 398 Degrees-F just prior to the time at which the first RCP was restarted (124 minutes after the reactor trip). Because the cold leg temperature decreased at a rate in excess of 100 Degrees-F per hour and the total temperature change of the cold leg was more than 100 Degrees-F, requirements for entry into the TSOR, based on Pressurized Thermal Shock, were met. The post-trip pressurizer level response remained on-scale. The Steam Generator pressure did not exceed a maximum of 995 psig. However, the pressures decreased to as low as 462 psig and 432 psig for SG "A" and "B" respectively (for the time period for which data are available).

The SG levels for SG "B" were within the range of 5% to 54% on the operating range. For SG "A", the levels reached 94%.

CAUSE OF EVENT

The first incident covered in this report is the fire in the 6900 V switchgear (1TA). Duke Power Design personnel and ITE-Gould manufacturing representatives examined the switchgear remains to attempt to establish an initial cause of the 1TA failure. It was possible to postulate theories of the switchgear failure such as arcing at "plug-in" connections which resulted in cross phase arcing, or a fire in the DC control circuitry cabling which caused cross phase arcing. However it was not possible to determine the cause with any certainty. Design Engineering and ITE-Gould personnel were able to determine that no counterfeit parts were evident. The breaker was not modified in any way and Preventative Maintenance had been performed per manufacturers recommendations. Due to the inability to determine a cause, this event is classified as Other.

The second incident is the reactor vessel overcooling. The Control Room Operators (CRO) were attempting to turn back the Reactor Coolant System (RCS) pressure increase without the ability to use pressurizer spray capabilities. However, the actions which they were taking, such as increasing RCS Makeup, were causing the RCS pressure increases which they were attempting to correct using Turbine Bypass Valves. The result was a decrease in cold leg temperature. The primary concern to the operators at the time of the event was the condition of the core. CRO "A", "B" and "C" were monitoring temperature through use of the Incore Thermocouples (ICTC) and attempting to achieve natural circulation cooling of the core. The ICTC were showing a steady temperature which is the indication which was of primary concern to the operators in determining core conditions. The operators were inadequately monitoring cold leg temperatures and therefore were not aware that the vessel was being subjected to an overcooling transient. The low decay heat content of the core due to the previous days trip contributed to the rapid cooldown. Had the event happened with the unit at 100% reactor power, the decay heat of the core would have probably prevented the rapid cooldown of the reactor coolant system and enhanced the establishment of natural circulation. Another error which occurred was the failure of the CROs to isolate FDW from the main nozzles. Since FDW was feeding through both the main and auxiliary nozzles, this contributed to the severity of the overcooling. It is recognized that the burden, including smoke in the control room, on the CROs at this time was not inconsequential since they were having to deal with a fire, ICS failures caused by fire damaged cables, and other problems. However the operators are trained on what criteria to monitor when attempting to achieve natural circulation. This training states that the SG levels should be raised to 50% while monitoring RCS temperature to ensure that overcooling does not occur. It also addresses the concern that with low decay heat loads, the RCS could be

overcooled. Due to the above basis, this incident is classified as Personnel Error, Improper Action.

There were several contributing causes to the overcooling. The most significant contributing cause was the burden on the operators. The Control Room Operators were attempting to control the plant and mitigate equipment failures and failures caused by the fire. They also were aware that the fire fighting effort was ongoing and had to deal with the stress of this situation. The mechanical failure of the yoke bearing on the "A" SG block valve (1FDW-31) and the calibration drift for the FDW control valves were two of the failures which were not fire related. The calibration drift allowed the control valves to back off the seat when the valves were placed in "MANUAL". The operators attempted to stop this overfeed condition by closing 1FDW-31 which would not close due to the yoke bearing failure. As a result of these items, the "A" SG overfed and subsequently lead to the tripping of the Main Feedwater Pumps. Another equipment failure which occurred during this incident was the failure of the pushbutton for the "B" SG block valve. The "OPEN" pushbutton stuck which caused the valve to cycle twice. The valve was finally maintained closed by opening the breaker. The failure of the Integrated Control System was a direct result of the fire. The signal cables had been damaged and as a result did not give correct indication.

The third contributing cause to this incident was ineffective training. The operators had been trained on the need to ensure that overcooling did not occur when the SGs were being fed up and also on the concern that overcooling was a possibility under low decay heat conditions. However, it is not possible at this time to simulate a low decay heat scenario on the simulator. Even though the operators had received the instruction in classroom sections, none of the operators who were interviewed recalled the caution about overcooling being a concern in low decay heat conditions.

The fourth contributing cause was the lack of procedural guidance for the establishment of Natural Circulation and the manual swapping of FDW flow from the main nozzles to the auxiliary nozzles. If there had been a procedure for the operators to use, it is probable that FDW would have been swapped correctly and the overcooling would have been less severe. A procedure would also have given the operators a guide to follow while attempting to establish natural circulation rather than relying on memory.

The third event was the decision by the Technical Support Center (TSC) not to perform the required three hour TSOR soak once the reactor overcooling had occurred. This decision was made due to the uncertainty of the start point for the TSOR temperature change criteria. The Emergency Operating

Procedure (EOP) lists two criteria, one of which must be met, as requirements for operation in the TSOR. The criteria are:

A. RCS temperature less than 500 Degrees-F and all RCPS off and HPI in the Engineered Safeguards (ES) mode.

B. Cooldown rate exceeds or has exceeded 100 Degrees-F per hour and a 100 Degrees-F temperature change has occurred.

TSC personnel were not sure if the start point for the temperature change in Criteria B was 500 Degrees-F, as listed in Criteria A. Criteria B would not have been met if this start point would have been correct. Some personnel in the TSC also was under the assumption that "HPI in the ES mode" meant that all three HPI pumps were on and injecting through both headers. Two pumps injecting through one header were in operation at the time of the event which would not have met the assumed criteria for TSOR operation. In actuality, both criteria for operation in the TSOR had been met. This is based on the cold leg temperature drop from the start of the transient and the fact that the "ES mode" consisted of at least one pump injecting through one header at a flow rate greater than normal makeup capability. Due to the reasons listed above, this event is listed as Management Deficiency, Deficient Supervision.

A contributing cause was the lack of guidance provided by the EOP. The confusion which TSC personnel had concerning the criteria for TSOR operation was a result of the vagueness of the EOP instructions.

The fourth incident was the utilization of Auxiliary Pressurizer Spray (APS) by the CRO. Operations Coordinator (OC) "A" was attempting to determine the cause of the APS failure and had the shift personnel initiate a momentary spray as a means to determine if the problem had been found and repaired. The Technical Specification (TS) concerns with the temperature differential between the Pressurizer and the spray fluid had been raised but were not felt to be applicable since OC "A" did not realize that the Pressurizer temperature was elevated. Due to this reason, this incident is classified as,Management Deficiency, Deficient Supervision.

A review of Licensee Event Reports which occurred during the past year did not reveal any other incidents similar to this event. Therefore this event is not recurring.

The response of the Fire Brigade personnel was very good. The operators used a logical sequence to attempt to extinguish the fire as it had been taught in training. Electrical isolations were performed, response was quick, and the fire was not allowed to spread to other areas of the plant.

The Shift Supervisor also had taken necessary actions to call out backups for the Fire Brigade. In discussions with the Shift personnel, a concern was raised on the number of trained fire brigade personnel. During the investigation, it was noticed that the number of readily accessible fire brigade personnel was barely adequate for the situation. Due to this concern, it is felt that the number of trained fire brigade personnel should be evaluated and increased accordingly.

No radioactive material releases, radiation exposures or personnel injuries occurred as a result of this incident. Therefore the health and safety of the public was not compromised.

CORRECTIVE ACTIONS

Immediate corrective actions were as follows:

The Fire Brigade was dispatched to extinguish the fire;

The Reactor and remaining Reactor Coolant Pumps were manually tripped;

Control Room Operators took action to initiate the Integrated Control System functions which had failed to occur;

Emergency Feedwater Pumps started automatically upon loss of the Main Feedwater Pumps;

A Reactor Coolant Pump was restarted;

Subsequent corrective actions were as follows:

A three hour soak was performed at stable Reactor Coolant System conditions;

The unit was brought to cold shutdown;

Operations sent out a training package giving guidance on requirements for entering the Thermal Shock Operating Region;

Babcock and Wilcox performed an analysis on the Pressurizer Spray Nozzle steam generator and reactor vessel which concluded that the transient had had a negligible impact on the nozzle;

Instrument and Electrical ran procedure IP/0/B/325/1 (ICS Feedwater Control A Loop Main FW and Startup FW valve Calibration) to correct the calibration of the Main Feedwater Control valves;

The 6900 V Switchgear was replaced during the EOC-11 Refueling outage;

Operations could not determine the reason why 1HP-340 (Auxiliary Pressurizer Spray) valve was closed. Previous valve checklists document the valve open;

Planned corrective actions are as follows:

Operations will compile a Training Package which will list all lessons learned from this incident and will train all Licensed Operators, Nonlicensed Operators, and Staff personnel on this Training Package;

Operations will review the Emergency Operating Procedure to ensure the necessary guidance for determining what the "ES Mode" of operation is, requirements for operation in the Thermal Shock Operating Region, guidance for the establishment of natural circulation, and guidance for manual swapping of Feedwater from the main nozzles to the auxiliary nozzles.

Operations will review T.S. 3.1.2.1 with all personnel involved with the violation of the cooldown limits.

Operations will review T.S. 3.1.2.6 with all personnel involved with the violation of T.S. 3.1.2.6 (Pressurizer temperature differential less than 410 Degrees-F).

Operations Training will revise training incorporating lessons learned from these incidents as necessary.

Operations Training will incorporate this incident into the events which are reviewed in Operator Requalification Training on a continuing basis.

The Transmission Department will review the results of the investigation reports on the switchgear failure and revise Preventative Maintenance requirements as necessary.

Projects will take action to replace the pushbuttons for the Main Feedwater Block Valves.

Mechanical Maintenance will rebuild 1FDW-31 and perform a stroke test prior to unit startup.

Station Management will evaluate Fire Brigade staffing levels to determine if present levels are sufficient.

SAFETY ANALYSIS

B&W analyzed the effects of the transient on the Steam Generator and the Pressurizer nozzles and determined that there had been no significant degradation caused by the transient. The yoke bearing on 1FDW-31 will be replaced by Mechanical Maintenance and the valve will be stroke tested successfully prior to unit startup. Instrument and Electrical recalibrated the Main Feedwater Control Valves within the required tolerances. The switchgear was rebuilt and functionally verified. Instrument and Electrical verified that the fire damage to the cables had been more than sufficient to cause the false indications to the Integrated Control System which resulted in the Feedwater anomalies.

The immediate response to the 1TA switchgear fire, the subsequent turbine trip and decision to trip the reactor did not pose any significant safety concerns. The reactor properly ran back from about 26% power to about 14% power upon the turbine trip. Reactor operation with two Reactor Coolant Pumps (RCP) is permissible and does not create any unusual or unacceptable operational problems. The decision to trip the reactor was totally based on fire fighting considerations (i.e., the 1TB switchgear was in the vicinity and had to be deenergized before the fire could be extinguished with a water fog) and not based on a threat created by fire on the reactor.

The events of potential concern with respect to safety are those subsequent to the manual reactor trip. Due to all the RCPs not operating, an attempt to establish natural circulation was in progress. Initially, upon loss of reactor coolant pump operation, ICS is to automatically switch main Feedwater (FDW) flow from the lower Steam Generator (SG) feed nozzles to the auxiliary feed nozzles, but this did not occur because the fire in the 1TA cubicle allowed a false "RCP on" signal to the ICS. The operators promptly noted the situation and opened valves allowing flow to the auxiliary feed nozzles and began to feed the SG to about 50% on the operating range per guidelines to achieve natural circulation. The status of natural circulation flow immediately established (i.e., slug flow vs. continuous flow) cannot be assessed from the available data; however, the core was adequately cooled at all times based on Incore Thermocouple readings by the operators and the hot leg temperatures from the available data.

Concurrent with the attempt to establish natural circulation, the Reactor Coolant System (RCS) was being pressurized with high makeup flow that was initiated by the operators prior to the trip. The operators took this action because the pressurizer level was decreasing during the power reduction from about 15% to 4%. The maximum RCS pressure reached was 2395 psig and did not result in the opening of the pressurizer's power operated relief valve (PORV) or safety valves. At this point, makeup flow was

throttled and RCS pressure decreased. Subsequently, RCS pressure did not exceed 2300 psig.

With rising primary pressure and not having Auxiliary Pressurizer Spray because of valve 1HP-340 being closed, operators opened the turbine bypass valves which initiated a rapid depressurization of the SG (to about 550 psig), cooldown, and depressurization of the RCS. Concurrent with the depressurizing of the SG, the operators were attempting to align main feedwater through the auxiliary feed nozzles with the intent of filling the SG to about 50% on the operating range. SG levels reached 60% in the "A" SG and 55% in the "B" SG in about 5 minutes. As a result of these actions, cold leg temperatures approached 460$^{O}$F, down from 575$^{O}$F when the reactor was at 15% power and approximately 550$^{O}$F when the reactor was tripped.

After this initial rapid cooldown (120$^{O}$F in 30 minutes), further cooldown followed the overfeeding of the "A" SG due to leakage flow through the main FDW control valve and the inability to immediately close the main FDW block valve. The SG was filled enough to trip the main FDW pump on high SG level and actuate emergency FDW. During this further cooling, cold leg temperatures in the "A" steam generator approached 426 Degrees-F at 59 minutes after the trip. Subsequent to these conditions, while the operators were still attempting to establish natural circulation, cold leg temperatures slowly decreased to about 400 Degrees-F due to the turbine bypass valves being open about 10%, the high SG levels, the RCP seal inleakage, and normal RCS makeup flow.

Operation in the Thermal Shock Operating Region (TSOR) is required when:

a.The RCS temperature is less than 500 Degrees-F and all reactor coolant pumps are off and the High Pressure Injection (HPI) has operated in the Engineered Safeguards (ES) mode, or

b.The primary cooldown rate has exceeded 100 Degrees-F per hour and a 100 Degrees-F temperature change has occurred.

Post-event debate on the above has taken place on two points, operation of the HPI in the ES mode and the initial point from which the 100 Degrees-F temperature change is measured. The first criterion was listed in the Emergency Operating Procedure as requiring HPI system operation in the ES mode. However the intended basis for entry into the TSOR is meant to include situations when makeup flow to the RCS exceeds normal makeup capacity, regardless of the operating mode of the HPI system. Based on the understanding at the time of the event, the first criterion for TSOR operation was never met. However, post event investigation revealed that the intent was to include excessive makeup flows. This criterion was therefore satisfied at approximately 10 minutes after the reactor trip when reactor coolant cold leg temperatures decreased below 500 Degrees-F.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The second point of debate centers on if the initial temperature from which to measure the 100_F change should be 500 Degrees-F (the temperature referenced in criterion a of the TSOR entry guidelines) or 575 Degrees-F (the last steady cold leg temperatures before reactor power reduction). The latter stance is the realization that the primary system components were also experiencing a thermal transient during the power reduction from 15% to 4% at which time the reactor was tripped. Based on this viewpoint, the second criterion for TSOR operation was satisfied at about 17 minutes after the reactor trip. Using 500 Degrees-F as the reference, this criterion was never satisfied. In the second case, the pressurized thermal shock limit was exceeded. The exceeded limit was not recognized and at almost two hours after the reactor trip, the Technical Support Center determined that operation in the TSOR was not required. The decision was to perform a three hour soak at a stable RCS pressure and temperature after a RCP was restarted, then complete a normal cooldown to cold shutdown conditions. As a subsequent action Operations issued a training package which determined the starting point for cooldown rate calculations. This point is defined as 550 degrees F if the unit is at 100% reactor power. If the unit is below 100% reactor power the starting point is the last stable cold leg temperature prior to the transient.

The real safety issue that now remains is the effect of the thermal and pressure stresses on the reactor vessel. B&W has completed its evaluations. Initial reports conclude that the stresses were acceptable. The complete evaluation along with details is in the process of being transmitted to Duke Power Company. B&W also evaluated the thermal stresses on the pressurizer spray nozzle because Auxiliary Pressurizer Spray flow was established and Technical Specification 3.1.2.6 was violated due to the temperature difference between the pressurizer and the spray being greater than 410 Degrees-F. Again, B&W analysis concludes that the thermal stresses were acceptable. Design Engineering evaluated the effects on the pressurizer spray line and the auxiliary pressurizer spray line and determined that these lines should not exceed experiencing two additional transients of this nature. The evaluation is establishing inservice inspection points for future surveillance.

While this event did not require an Engineered Safeguards Feature actuation, operator response was definitely required to restore the plant to normal conditions. The maximum post-trip RCS temperatures were at the time of the trip, resulted in an average coolant temperature of 555 Degrees-F. The minimum average temperature for which hot leg temperature data is available is 490 Degrees-F. The RCS cold leg temperature in the 'A' loop decreased from about 550 Degrees-F at the time of the trip to 426 Degrees-F at 59 minutes after the reactor trip and fell to 398_F at 124 minutes after the reactor trip, just prior to the first RCP being restarted. The RCS pressure ranged from a minimum of 1995 psig to a maximum

of 2395 psig. The pressurizer level remained on-scale and reached a minimum of 110 inches and a maximum of 185 inches. The post-trip SG pressure did not exceed 991 psig in the "A" SG and 995 psig in the "B" SG, but the pressures did reach minimums of 462 psig and 432 psig, respectively (for the post-trip time period for which data is available). The "B" SG level ranged between 5% and 54% on the operating range, but due to the main FDW control valve's leakage flow and the main FDW block valve's failure to fully close, the "A" SG was overfed to the point of tripping the main FDW pumps. The maximum level reached in the "A" SG was 94% on the operating range.

Throughout this event, adequate core subcooling margin was maintained and fuel integrity was not threatened. In addition, the primary system integrity was not breached and a release of radioactivity did not occur. Based on these facts and the preceding analysis, the health and safety of the public were not affected.

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**DUKE POWER**

February 2, 1989

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

Subject: **Oconee Nuclear Station**
**Docket Nos. 50-269, -270, -287**
**LER 269/89-02**

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report (LER) 269/89-02 concerning a fire in 1TA switchgear.

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PJN/ler5

Attachment

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