AEOD TECHNICAL REVIEW REPORT*

UNIT: La Salle County Station Unit 1
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NSSS/AE: General Electric/Stone & Webster

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SUBJECT: EXCESSIVE COOLDOWN RATE EVENT AT LA SALLE UNIT 1

SUMMARY

On July 18, 1983, with La Salle Unit 1 in startup, all turbine bypass valves opened suddenly due to work being done on the electro-hydraulic control (EHC) system. Following the turbine bypass valves opening the motor-driven reactor feed pump tripped on vessel high level due to swell. When power was restored to the electro-hydraulic control system, the bypass valves closed, and reactor pressure vessel level stabilized at approximately -40". The bypass valves once again opened due to electro-hydraulic control card movement and vessel level dropped to -50". This caused initiation of the primary containment isolation system, the high pressure core spray system and reactor core isolation cooling system. Vessel water level then recovered. The combined effects of cold water injection and coolant flashing from the depressurization resulted in a cooldown rate of 113°F/hr.

The 113°F/hr cooldown rate during the event was larger than the maximum cooldown rate of 100°F/hr allowed by the Technical Specifications. To assess the effects of this high cooldown rate, General Electric Company performed an engineering evaluation for the licensee. The evaluation concluded that the cooldown rate, although slightly in excess of the limits, had no significant structural effects on the reactor pressure vessel and that the nil ductility limit had not been reached. Therefore, this event did not involve significant safety consequences and did not result in reactor pressure vessel structural limit being exceeded.

*This document supports ongoing AEOD and NRC activities and does not represent the position or requirements of the responsible NRC program office.

DISCUSSION

On July 18, 1983, with the La Salle Unit 1 plant in startup, work was being done on the main turbine electro-hydraulic control (EHC) system. The reactor power level was 322 MWT at this time, which is about 10% of full rated power. During this time an instrument maintenance department technician removed a power supply card for the EHC system. This resulted in a loss of power to the EHC system which caused all five turbine bypass valves to open. The motor-driven reactor feed pump (MDRFP) tripped on high vessel water level from the swell caused by the vessel depressurization. With the bypass valves open, the pressure in the reactor vessel dropped for approximately 2.5 minutes. Power was then restored to the EHC system which caused the bypass valves to reclose. Reactor vessel level subsequently dropped to -40" after the initial swell. The reactor protection system (RPS) also performed properly as the control rods inserted when reactor water level fell to +12.5" corresponding to the "low" level scram set point.

Approximately 3.5 minutes after the level leveled off at -40", the bypass valves reopened and then reclosed again due to another EHC power supply card movement. This sequence caused the level to drop to -50". At the -50" level, the high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems automatically initiated. Initiation of the HPCS and RCIC systems restored reactor pressure vessel (RPV) water level. Several primary containment isolation system (PCIS) groups also inititated when reactor water level dropped to -50". All isolation valves closed properly with the exception of valve 1CM027. Valve 1CM027 is a drywell air sampling line isolation valve. It indicated open and failed to close with the isolation signal present. The valve was closed manually.

The combined effect of HPCS and RCIC injection, together with the coolant flashing, caused by the depressurization, resulted in a cooldown rate of 113°F/hr. This cooldown rate exceeded the 100°F/hr cooldown rate allowed by the Technical Specifications.

An investigation of this event was conducted to review:

1. The plant protection systems responses.

2. The cooldown rate which exceeded 100°F/hr.

3. The failure of valve 1CM027 to close.

4. The EHC transient relative to the plant safety analysis basis.

During the cooldown incident, all of the reactor protection systems functioned properly except valve 1CM027 which failed to close. The MDRFP tripped when the high level set point was reached during the swell. The control rods scrammed at low reactor water level of +12.5". When the RPV level stabilized at -50", HPCS and RCIC systems were initiated and reactor water level recovered.

The cold water addition to the RPV and the excessive depressurization rate resulted in a cooldown in excess of 100°F/hr and resulted in a violation of the Technical Specification limit. Technical Specification 3.4.6.1 states that reactor coolant temperature shall not exceed maximum cooldown rate of 100°F/hr. In this incident a cooldown rate of 113°F/hr occurred due to the opening and closing of turbine bypass valves during EHC troubleshooting.

To evaluate the consequences of the Technical Specification violation, the General Electrical Company performed an engineering evaluation. The following data was supplied by the licensee to the General Electric Company (GE) for the analysis:

1. Cooldown rate as calculated from steam space pressure and temperature of approximately 113°F/hr.

2. Maximum metal temperature change of 10°F during the transient.

3. Reactor vessel level remained below shell flange.

The GE evaluation led to the following results:

1. Reactor pressure vessel flange bolts are the limiting component.

2. The total fatigue usage for the limiting component was not affected as determined by the GE analysis.

3. The consequence of this event was less severe than a normal shutdown during which water quenching of the flange occurs due to flooding.

Thus, the engineering evaluation performed by GE concluded that no structural effects on the reactor vessel had occurred and that no nil ductility limits had been approached.

The failure of valve 1CM027 to close could not be repeated by the licensee. The valve when tested functioned properly when the proper PCIS signals were simulated and the control switch cycled. A reason for the failure of valve 1CM027 to close could not be determined by the licensee.

An attempt was made to compare the consequences of this transient to the limiting EHC transient discussed in the plant safety analysis. A pressure regulator failure which caused the turbine bypass valve to open as reported in the FSAR is the transient most similar to the EHC transient. However, the pressure regulator failure analyzed in the FSAR is from full power conditions whereas the EHC transient event analyzed here had an initial power level corresponding to approximately 10% of the full power.

During reactor operation, a malfunction of the EHC system pressure regulator could cause a low steam pressure condition in the reactor and at the turbine inlet if the turbine control valves and/or turbine bypass valves were to fully open. From partial power operations, if unchecked, the rate of nuclear system saturation temperature reduction could exceed the Technical Specification cooldown rate limit. To avoid a violation of the Technical Specification cooldown rate caused by a pressure regulator malfunction, a Group 1 isolation signal on low steam header pressure is provided to isolate: (1) all four main steam lines, (2) the main steam line drain lines and (3) the reactor water sample line. However, in this instance the installed automatic protective arrangements were not capable of preventing the cooldown rate from being exceeded following the EHC malfunction in which the power supply card was withdrawn by the instrument maintenance department technician.

The inability of the installed protective arrangements to prevent an excessive cooldown rate in this event was a result of a system perturbation beyond the design basis of the installed protective arrangements. The design basis for the protective equipment is a pressure regulator failure which causes the bypass valves to open and remain open. For such an event a monotonic decrease in reactor vessel pressure and temperature results which is arrested by a Group 1 isolation before the emergency core cooling systems are started automatically. In such an event, the timeliness of the vessel isolation is sufficient to prevent an excessive cooldown rate. For the EHC malfunction in the La Salle event over a few minute period, the bypass valves were opened, closed, and then reopened and reclosed. The resulting pressure and level perturbations to the reactor coolant system caused the HPCS and RCIC systems to be actuated on low water level. The attendant cooldown rate caused by the combined effects of (1) bypass steam flow, (2) RCIC cold water injection, (3) HPCI cold water injection and (4) RCIC turbine steam flow was sufficient to cause the cooldown rate to exceed the Technical Specification limit. That is the system transient which occurred as a result of the sequential EHC malfunctions caused by the repetitive power supply card movements and was beyond the design basis for the installed protective equipment.

However, the installed protective equipment is intended to prevent an unwanted violation of a reactor vessel design limit rather than a reactor safety limit. Thus, the failure of the installed protective equipment to prevent the violation of the Technical Specification cooldown rate (design) limit is of relatively minor safety significance. Furthermore, the inability of the protective arrangements to prevent violation of Technical Specification limit for the relatively unique event which occurred at La Salle is not considered to be an adequate basis for reassessing the design basis for the existing protective arrangements.

To avoid this type of failure in the future, the licensee will give additional indepth training on the EHC system to the instrument maintenance personnel.

FINDINGS

1. All of the emergency safety functions operated properly except valve 1CM027 which failed to close.

2. An instrument maintenance department technician reportedly pulled a power supply card during EHC troubleshooting which led to a plant disturbance which exceeded the design basis for the installed protective equipment.

3. GE analysis indicated the nil ductility limit was not approached and no significant structural effects on the reactor vessel had occurred during the 113°F/hr cooldown rate.

CONCLUSIONS

The cause of this incident was a personnel error involving a misinterpretation of the effects of withdrawing a power supply card during EHC troubleshooting. The licensee estimated that the cooldown rate during this transient was 113°F/hr which is above the limit allowed by the Technical Specification. However, an evaluation of the consequences by GE indicated that no structural effects on the reactor vessel had occurred and that no nil ductility limits had been approached. The failure of valve 1CM027 to close could not be duplicated by the licensee. The valve when tested functioned flawlessly when PCIS signals were simulated and the control switch cycled. The licensee also could not determine the reason for the failure of valve 1CM027.

The plant had protection installed to prevent a plant cooldown rate which might exceed the technical specification limit; however, this limit was violated as a result of a human error which led to a plant transient which exceeded the design basis for the equipment. Because of the uniqueness of the disturbance, the existing design of the protective arrangements are still considered to be adequate. The licensee has stated that they have taken steps to avoid this type of failure in the future by retraining their instrument maintenance personnel. We consider the actions to be sufficient to address this event.

REFERENCES

1. LER 83-084/03L-0

2. LER 83-085/03L-0

3. LER 83-087/03L-0

4. DVR 1-1-83-270 GE Engineering Evaluation of August 19, 1983.

5. Conversations with La Salle Resident Inspector, January 1984.