NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| BYRON NUCLEAR POWER STATION, UNIT 1 | 05000454 | 1 OF 5 |

TITLE (4)

DEPRESSING BOTH FEEDWATER ISOLATION RESET PUSHBUTTONS LEADS TO LCO 3.0.3 ENTRY

| 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 NAME | DOCKET NUMBER |
| 04 | 22 | 99 | 99 | -- 001 -- | 00 | 05 | 21 | 99 | FACILITY NAME | DOCKET NUMBER |

| OPERATING MODE (9) | 3 |
|---|---|
| POWER LEVEL (10) | 0% |

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

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

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Jose Dubon | 815-234-5441 x4504 |
| Paul Geddes | 815-234-5441 x4082 |

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 |
|---|---|---|---|---|---|---|---|---|---|---|
| N/A | | | | | | | | | | |
| | | | | | | | | | | |

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

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

On 4/22/99, during control rod drop testing in Mode 3, prior to reactor startup following refueling (B1R09), the reactor trip breakers were cycled numerous times. The reactor trip permissive interlock (P-4) signal associated with the trip breakers opening results in a feedwater isolation signal. To prevent this unwanted feedwater isolation, Byron Station's past practice had been to temporarily depress and hold feedwater isolation reset pushbuttons while opening the reactor trip breakers. This action blocks both trains of feedwater isolation from occurring during the P-4 interlock actuation, and also effectively renders 2 trains of feedwater isolation actuation instrumentation inoperable, because a feedwater isolation can not occur during the time the pushbuttons are depressed.

Prior to implementation of the Improved Standard Technical Specifications (ITS), the feedwater isolation function was only required in Modes 1 and 2, and thus rendering feedwater isolation actuation instrumentation inoperable during control rod drop testing was not an issue. After implementation of ITS, the feedwater isolation actuation instrumentation function is required in Modes 1 through 3, and rendering 2 trains inoperable in Mode 3 resulted in an entry into Limiting Condition for Operation (LCO) 3.0.3. This was determined to be a condition prohibited by the Technical Specifications, reportable under 10CFR50.73(a)(2)(i)(B).

The cause of the event was determined to be that the procedure review before implementing ITS did not make the necessary changes to this procedure. It was determined that there was minimal safety significance to the event.

Procedures will be changed to use alternative testing methods that avoid the LCO 3.0.3 entry.

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time: 4/22/99 / 0800

Unit 1 Mode- 3 - Rx Power 0% RCS [AB] Temperature/Pressure 557/2235

Unit 2 Mode - 1 - Rx Power 100% RCS [AB] Temperature/Pressure NOT/NOP

B. DESCRIPTION OF EVENT:

Byron Station Unit 1 was completing the ninth refueling outage. During control rod drop testing while in Hot Standby (Mode 3), just prior to reactor startup following refueling, the reactor trip breakers were cycled numerous times by procedure. The specific testing consists of withdrawing a bank of control or shutdown rods, and then pulling the stationary gripper coil fuses one at a time, while a recorder trace times the individual rod falling into the core. The rod drive cabinets must be energized during fuse withdrawal in order for the rod drop test to work correctly. However, during reinsertion of the fuses after a test section is completed, it is desirable for personnel safety reasons to have the rod drive cabinets de-energized by opening the reactor trip breakers. This prevents any inadvertent contact with energized components during fuse reinstallation.

Opening the reactor trip breakers results in a reactor trip permissive interlock (P-4) signal, which has multiple functions/effects, one of which is cause a feedwater isolation. Since Unit 1 was in Mode 3 at normal pressure and temperature, a small amount of feedwater, approximately 20 gpm, was flowing to each steam generator in order to maintain reactor coolant system temperature. Feedwater isolation is undesirable in this condition, because it results in a thermal cycle on the feedwater nozzles, which are the feedwater line penetrations into the shell of the steam generators, and is an unnecessary actuation of Engineered Safety Feature equipment.

To prevent this unwanted feedwater isolation, Byron Station's past practice had been to temporarily depress and hold feedwater isolation reset pushbuttons while opening the reactor trip breakers. This practice was allowed by the Technical Specifications in effect before the implementation of the Improved Standard Technical Specifications, because feedwater isolation actuation instrumentation [JE] was not required to be operable until Startup (Mode 2). The Improved Standard Technical Specifications has a more restrictive requirement in that feedwater isolation actuation instrumentation is now also required to be operable in Mode 3, except when all feedwater isolation valves are closed, or isolated by a closed manual valve. The Improved Standard Technical Specifications were implemented at Byron Station on February 6, 1999. The site process for implementing the Improved Standard Technical Specifications failed to recognize that this testing technique would result in noncompliance with Technical Specification requirements.

B. DESCRIPTION OF EVENT (cont.)

During the test, the feedwater isolation reset pushbuttons only need to be held in for a few seconds. This action blocks both trains of feedwater isolation from occurring during the P-4 interlock actuation. However, it also effectively renders 2 trains of feedwater isolation actuation instrumentation inoperable, because a feedwater isolation can not occur during the time the pushbuttons are depressed. Holding in the reset pushbuttons prevents the slave relay from energizing. The investigation revealed that the only portion of the feedwater isolation actuation instrumentation affected was the reactor trip actuation of feedwater isolation. The other two feedwater isolation actuation signals, safety injection and steam generator high-high level (P-14), would have worked as required even with the reset pushbuttons depressed. Additionally, the feedwater isolation valves remained operable. At all times during this event the feedwater isolation valves could have been manually closed. However, the definition of OPERABILITY requires that all parts of a circuit be functional in order for the circuit to be considered OPERABLE. Therefore, the determination was made that the TS requirement in Table 3.3.2-1 function 5a was not met and entry into LCO 3.0.3 was required.

The Operations Shift Manager determined that Byron Station Unit 1 had inadvertently entered LCO 3.0.3 several times on 4/22/99. It was later determined that this situation was reportable. Subsequently, Station Management convened to discuss the appropriate actions to allow continuation of scheduled surveillances (e.g. control rod drop testing). Several options were explored in conjunction with Corporate Nuclear Generation Group (NGG) Management. These options included isolating feedwater flow from the steam generators for the duration of the rod drop testing; allowing the feedwater isolation signal to actuate approximately 10 times to facilitate completion of testing; pursuit of enforcement discretion with the NRC, or providing direction to the Operations shift to continue testing in the same manner as before via continued short-term entry in LCO 3.0.3. The decision was made by Station Management to pursue testing in the same manner as previous via short-term entry into LCO 3.0.3. This decision was based on a review of the technical concerns related to the isolation options, and the determination that performing the testing by temporarily blocking the feedwater isolation circuitry was the best option, and had minimal safety impact. The primary functions of the feedwater isolation signal is to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the steam generators. This function is necessary to mitigate the effects of a high water level in the steam generators, which could result in carryover of water into the steam lines, and excessive cooldown of the primary system [AB]. Thus, the feedwater isolation Engineered Safety Feature function from P-4 provides minimal safety benefit in Mode 3 conditions with the main turbine offline, and very low feedwater flows to the steam generators. In addition, discussions were held between representatives of ComEd and the NRC to notify the NRC of Byron Station's intent to continue testing by temporarily defeating the feedwater isolation circuitry when opening the reactor trip breakers.

C. CAUSE OF EVENT:

The cause of the event was determined to be an implementation error in the procedure review for conversion to the Improved Standard Technical Specifications. During the review process, it was not recognized that blocking both trains of feedwater isolation actuation instrumentation in Mode 3 was now a condition prohibited by the new Technical Specifications.

D. SAFETY ANALYSIS:

The primary functions of the feedwater isolation signal are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the steam generators. These functions are necessary to mitigate the effects of a high water level in the steam generators, which could result in carryover of water into the steam lines, and excessive cooldown of the primary system. The turbine was offline during this event, and is always maintained offline during Mode 3. Therefore preventing damage to the main turbine was not a concern during this event. Feedwater flow to the steam generators was throttled to a very low value, approximately 20 gpm per steam generator, and thus the likelihood of an excessive cooldown due to feedwater flow was very low. The feedwater isolation valves remained operable at all times, and could have been closed at any time had a concern about excessive cooldown arisen. Furthermore, in the unlikely event a Safety Injection had occurred during the time the reset pushbuttons were being pushed, feedwater isolation would have occurred as required. In the unlikely event a P-14 high-high Steam Generator level signal had been received during the time the reset pushbuttons were being pushed, feedwater isolation would have occurred as required because only the P-4 feedwater isolation function was affected, as stated earlier. The Technical Specifications Bases section discussing P-4 states that none of the functions associated with P-4 (feedwater isolation included) serves a mitigation function in the plant licensing basis safety analyses. It also states that feedwater isolation is not required to show that the plant licensing basis safety analysis acceptance criteria are met. Thus, this event had no impact on public health and safety.

E. CORRECTIVE ACTIONS:

A review will be conducted of the procedures affected by changes identified as "more restrictive" during the conversion to the Improved Standard Technical Specifications. This review will ensure that all other procedure changes required for the implementation of the Improved Technical Specifications were made.

At the time of the event, alternate means of performing the Rod Drop Surveillance Testing that would not involve an entry into LCO 3.0.3 were sought. No other means were readily apparent. Subsequent investigation revealed that there was an acceptable alternative. Both reactor trip breakers did not need to be tripped in order to satisfy the personnel safety concern. One reactor trip breaker could have been tripped locally, while momentarily depressing the associated train of feedwater isolation reset pushbutton. This would have prevented the feedwater isolation, while only rendering one train of feedwater isolation inoperable. The rod drive cabinets would have been de-energized, allowing safe reinstallation of fuses. With one train of feedwater isolation inoperable, entry into LCO 3.3.2 Condition G is required, which requires restoration of the inoperable train within 6 hours. Once the reset pushbutton is released, however, operability is restored to the train of feedwater isolation, and Condition G is exited. No LCO 3.0.3 entry is required for this method, and the total amount of time in Condition G is only a few seconds.

It was also discovered that the concern about thermal cycles on the feedwater nozzles was too restrictive. The steam generators are designed for up to 2000 cycles of feedwater isolation while in Mode 3 during a 40-year lifetime, assuming a feedwater inlet temperature of 32 degrees F. Actual feedwater temperature during this event was 85 degrees F.

The rod drop testing procedure will be changed to utilize alternative methods that avoid LCO 3.0.3 entry. A review will be conducted to determine if there are any other similar occurrences, and changes will be made to utilize alternative methods that avoid LCO 3.0.3 entry.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

No other instances of errors associated with the implementation of the Improved Standard Technical Specifications have occurred at Byron. A review of similar events at other utilities was conducted. None were found that applied.

G. COMPONENT FAILURE DATA:

None