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

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1): | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Braidwood Unit 1 | 05000456 | 1 of 7 |

TITLE (4) Potential Failure of Westinghouse Fuel Rod Design Criteria discovered by Westinghouse while updating PAD code.

| 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 |
| | | | | | | | | | Braidwood Unit 2 | 05000457 |
| 10 | 28 | 97 | 97 | 009 | 01 | 3 | 19 | 98 | FACILITY NAME | DOCKET NUMBER |

| | |
|---|---|
| OPERATING MODE (9) | 1 |
| POWER LEVEL (10) | 99.9 |

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

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

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Lonnie Kepley, Nuclear Group Leader, System Engineering | (815) 458-2801 Extension 2490 |

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 |
|---|---|---|---|---|---|---|---|---|---|---|
| None | | | | | | | | | | |
| | | | | | | | | | | |

| 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 fifteen single-space typewritten lines 16)


Westinghouse analysis shows that for high power, high burnup assembly Integral Fuel Burnable Absorber fuel rods, calculated rod internal pressure are in excess of the Westinghouse fuel rod design criterion that "the fuel rod pellet-to-clad gap shall not re-open." Additionally, if the "no gap re-opening" criterion is exceeded, the 17% clad oxidation limit following postulated Loss of Coolant Accident event, as defined in 10CFR50.46, can likewise be exceeded. Westinghouse has since determined that all currently operating plants are in compliance with 10CFR50.46.

Cause of the issue was Westinghouse used inadequate methods and incorrect assumptions in fuel rod design code.

Corrective actions: Westinghouse completed plant-specific analyses on March 6, 1998. These analyses indicate that gap re-opening occurs during Braidwood 1 Cycle 7 operation, but that the 10 CFR 50.46 cladding oxidation criterion continues to be met. The criterion will be met through the cycle's End of Life (EOL) burnup of 20,000 MWD/MTU. In addition, all analyses supporting the conclusions of the Cycle 7 Reload Safety Evaluation demonstrate that operation until EOL is valid. Braidwood System Engineering and Nuclear Fuel Services have reviewed the results of these analyses and concur with the conclusion.

A search of the Nuclear Station Regulatory Assurance database found no previous occurrences.

The fuel rod design issues have the potential to have the nuclear fuel outside the design basis according to 10CFR72(a)(2)(ii).

A. **PLANT CONDITIONS PRIOR TO EVENT:**

Unit(s): 1 Event Date: 10/28/97 Event Time: 1230 Hours
Reactor Mode(s): 1 Power Level(s): 99.9% RCS [AB] Temp./Press. NOT/ NOP

B. **DESCRIPTION OF EVENT:**

There were no systems or components inoperable at the beginning of this event that contributed to the severity of the event.

Background:

Westinghouse Electric Company's fuel performance code, Performance Analysis and Design (PAD), is an Nuclear Regulatory Commission (NRC) approved methodology that has been in use since the 1970s as part of nuclear plants' licensing basis. Westinghouse uses PAD to demonstrate that fuel performance criteria to which plants are licensed are met for each reload core and to provide reference fuel temperature inputs to various safety analyses.

In early 1996, Westinghouse discovered that the rod internal pressure buildup due to helium release from Integral Fuel Burnable Absorbers (IFBA) was higher than previously modeled. Measurements taken during the hot cell examination of fuel rods indicated that more of the helium was contributing to pressure in the gap/plenum regions of the fuel rods and less was being retained in the fuel/coating material matrix than previously measured. Because of the few data points available, Westinghouse has conservatively assumed 100% theoretical helium release in all IFBA rods.

In late 1996, Westinghouse completed development of a new corrosion model for Zirc-4 cladding material to address the higher levels of corrosion being measured in the field on high duty fuel rods. This new corrosion model was presented to the NRC in December 1996. Westinghouse has been pursuing the incorporation of this model into PAD and assessing the feedback effects on other fuel performance criteria. During this time, Westinghouse has been applying the revised Zirc-4 corrosion model, without incorporation into PAD, to all core designs to confirm less than steady-state oxide accumulation will meet the design criteria.

After incorporation of the new corrosion model, PAD indicates that the higher levels of corrosion are causing elevated fuel cladding temperature at end of life (EOL) conditions, and consequently, higher outward clad creep rates and reduced pressure margin to the no gap re-opening criteria. Westinghouse believes that conservatism exists in the PAD code that will compensate for the increased corrosion feedback effect.

B. **DESCRIPTION OF EVENT:** (continued)

Current Events:

On 10-28-97 at 1230, Westinghouse Electric Company reported to ComEd Braidwood Nuclear Station that current modeling for fuel design demonstrates there is a potential to not meet the 10CFR50.46 design criteria for maximum cladding oxidation of 17% for IFBA fuel which is in the second cycle of operation.

Normally, fuel is used in the reactor for three cycles with each cycle lasting approximately eighteen months. Certain operating fuel regions may potentially exceed the gap reopening design criterion during the second half of the second cycle. If the gap reopening design criterion is exceeded, the 17% local oxidation criterion following a postulated Loss of Coolant Accident (LOCA) may likewise be exceeded. This gap reopening has the potential to have the nuclear fuel outside the design basis and reportable according to 10CFR72(a)(2)(ii). This issue was reported by Braidwood Station to the NRC Operations Center via the Emergency Notification System at 1235 on 10/28/97.

This concern was reported by Braidwood Nuclear Station as a condition potentially outside the design basis of the plant. Westinghouse reported this as a generic issue to the NRC and supplied information for a generic Justification for Continued Operation (JCO) for all plants, including Braidwood Station. In the JCO, Westinghouse concluded that a substantial safety hazard does not exist for operating plants. The basis for this determination is that shutdown capability will not be hindered, coolable core geometry will be maintained to the extent required in all design basis accidents and 10CFR100 offsite dose limits will be met since additional fuel failures are not expected.

As documented in Braidwood Station's Operability Assessment, dated 10-29-97, ComEd Nuclear Fuel Services (NFS), Westinghouse, Braidwood Engineering and Byron Engineering have determined that there is reasonable assurance that the 10CFR50.46 criteria for maximum allowable oxidation of fuel cladding for the two operating Braidwood reactor cores is not exceeded.

B. **DESCRIPTION OF EVENT: (continued)**

Current Events:

On 11-6-97, the Westinghouse Owners Group (WOG) and Westinghouse representatives met with the NRC to provide a technical update on the issue. Representatives from a number of utilities including ComEd attended the meeting. One of the NRC's top concerns was compliance with the 10CFR50.46 criteria (17% post-LOCA clad oxidation limit). At the meeting, a screening process was presented to the NRC to address this concern. The concern is limited to plants with IFBA fuel during the second half of the second cycle of operation. Based on preliminary calculations, if fuel has less than 12% cladding oxidation of corrosion there is no potential to exceed the 50.46 criteria. The limiting plant has been evaluated and will not reach this point until approximately 12-25-97. Therefore, as of 11-6-97 all operating plants are in compliance with the 50.46 criteria, with the possible exception of Byron Unit 1. The WOG and Westinghouse committed to providing the NRC, with a short list of affected plants, and approximate dates when corrosion levels will exceed 12% cladding oxidation by the end of the first quarter of 1998. This list sets the priority for plant specific evaluations and analyses to demonstrate compliance. Both Braidwood Units are on this short list with estimated dates of 3/1/98 and 7/1/98, respectively, to exceed 12% cladding oxidation.

This event is being reported pursuant to 10CFR50.73(a)(2)(ii).

Westinghouse completed additional plant-specific analyses on March 6, 1998. These analyses indicate that gap re-opening occurs during Braidwood Unit 1 Cycle 7 operation, but that the 10 CFR 50.46 cladding oxidation criterion continues to be met. The criterion will be met though the cycle's End of Life (EOL) burnup of 20,000 MWD/MTU. In addition, all analyses supporting the conclusions of the Cycle 7 Reload Safety Evaluation demonstrate that operation until EOL is valid. Braidwood System Engineering and Nuclear Fuel Services have reviewed the results of these analyses and concur with the conclusion.

C. **CAUSE OF EVENT:**

Westinghouse initiated a program to obtain additional detailed performance data on its fuel. Many IFBA rods were sent to the AECL hot cell and corrosion data was obtained from various sites. This hot cell data showed that the helium release from IFBA rods was greater than what had been previously measured. Specifically:

- IFBA Helium release was shown by test data to be greater than predicted by the model. This resulted in a change to the Helium release model to conservatively assume 100% He release.

C. **CAUSE OF EVENT: (continued)**

- Corrosion on high duty fuel rods was shown by test data to be greater than predicted by the model. This resulted in a new corrosion model being incorporated into PAD.

- When the corrosion model is used, PAD results indicate that the higher levels of corrosion lead to higher fuel cladding temperatures, which result in higher outward clad creep rates.

- The revised helium release model results in higher fuel rod internal pressures, which also results in higher outward clad creep rates.

- When the revised models are applied to a bounding fuel rod duty scenario, i.e., the worst case combination of conservative assumptions, results show that the gap between the pellet and fuel rod may reopen.

- Gap reopening is in violation of a Westinghouse fuel rod design acceptance criterion, and, in some fuel rods, may lead to >17% clad oxidation during LOCA conditions.

Westinghouse has been updating their PAD code to analyze the fuel and determined, using generic plant data, that the 17% cladding oxidation has a potential of being exceeded.

D. **ASSESSMENT OF SAFETY CONSEQUENCES:**

Gap Re-opening:

Westinghouse has performed a comprehensive safety assessment, which considered the potential for gap reopening in operating cores. When considering the impacts of gap reopening on fuel reliability, two outcomes can be postulated: (1) no rod failure, and (2) rapid rod failure. A third postulated scenario (slow rod failure) was determined to not be credible, because no mechanism could be postulated which would lead to slow failure. The no failure outcome is concluded to be the most probable outcome. This leads Westinghouse to conclude that gap reopening will not lead to fuel rod failure. Since gap re-opening does not lead to fuel rod failures and since previously analyzed design basis accident scenarios remain bounding, Westinghouse concludes that gap reopening is of low safety significance.

D. **ASSESSMENT OF SAFETY CONSEQUENCES:** (continued)

17% Cladding Oxidation:

Westinghouse has an initial 12% oxidation criteria to permit assessment of plants regarding compliance with the 17% maximum cladding oxidation criteria of 10CFR50.46. As additional sensitivity analyses are performed, the screening criteria pre-oxidation value may be increased from the 12% level. Based on the screening criteria, plants with integral burnable absorber fuel in the first half of their operating cycle or returning from a refueling outage are in compliance due to either no gap re-opening or low levels of pre-oxidation due to steady-state corrosion accumulation. For plants with integral burnable absorber fuel in the latter half of their operating cycle, gap re-opening may occur. All plants that are predicted to have fuel rod gap re-opening, but which have pre-oxidation of the clad less than approximately 12%, are in compliance. The basis for this conclusion is the LOCA sensitivity studies, which showed that the oxidation resulting from a LOCA is less than 3% under bounding conditions, as applied to burned fuel, assuming gap reopening. Therefore, the sum of the pre-transient and transient oxidation from the studies performed has been shown to be less than the 17% criteria, given initial oxidation levels less than approximately 12%. Westinghouse considers the 12% initial oxidation value to be conservative.

As of November 6, 1997, given the available information which needs to be verified all operating plants pass the screening criteria and are therefore in compliance.

During this entire sequence of events, there were no adverse consequences to the health and safety of the general public or plant personnel as a result of the potential of not meeting the 10CFR50.46 design criteria for maximum cladding oxidation of 17% for fuel during the second cycle of IFBA fuel in the reactor.

E. **CORRECTIVE ACTIONS:**

Immediate Actions:

Westinghouse identified the potential of not meeting the 10CFR50.46 design criteria for maximum cladding oxidation of 17% for fuel during the second cycle of IFBA fuel and notified licensees and the NRC.

Corrective Actions:

Westinghouse has developed a comprehensive plan to resolve the fuel rod internal pressure issue. This plan has three steps: (1) review and improvement of analytical models, (2) gathering of additional data, and (3) performance of plant by plant assessments.

**E. CORRECTIVE ACTIONS: (continued)**

Westinghouse believes that conservatism exists in the PAD code that will compensate for the increased corrosion feedback effects. These conservatisms include the use of unirradiated cladding creep rates, a pellet-to-clad contact model which is restricting the axial growth of the fuel rod, and a strain reversal algorithm which conservatively predicts the point of gap re-opening. These effects need to be incorporated in the PAD model and appropriately verified.

Prior to introduction of these modeling improvements, a detailed review of the current PAD methods and models will be performed. While the analytical activities are underway, collection of additional field data will be pursued to validate fuel performance methods. Upon completion of the PAD development, a plant-by-plant assessment will be performed with the modified PAD model, using plant specific conditions.

Braidwood Nuclear Engineering will monitor and evaluate the Braidwood assessment scheduled to be received from Westinghouse in mid-February. (NTS Number 456-180-97-SCAQ00009-01)

**This action item has been completed.**

Braidwood Station will submit a supplement to this LER based on the results of the plant specific assessment. (NTS Number 456-180-97-SCAQ00009SR-01)

**This action item has been completed.**

An effectiveness review of any corrective actions documented in this supplement will be scheduled in accordance with station procedures and documented in the supplement. (NTS Number 456-180-97-SCAQ00009ER)

**F. PREVIOUS OCCURRENCES:**

Searched Nuclear Station Regulatory Assurance (ALRA) database using key words "Westinghouse", "fuel", "design" and "criterion" and found no occurrences to indicate this has happened at ComEd in the past.

**G. COMPONENT FAILURE DATA:**

MANUFACTURER -----NOMENCLATURE MODEL MFG. PART NO.

Since no component failure occurred, this section is not applicable.