Commonwealth Edison Company
Byron Generating Station
4450 North German Church Road
Byron, IL 61010-9794
Tel 815 234-5441

ComEd

November 24, 1997

LTR: BYRON 97-0280
FILE: 3.03.0800 (1.10.0101)

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

Dear Sir:

The Enclosed Licensee Event Report from Byron Generating Station is being transmitted to you in accordance with the requirements of 10CFR50.73(b)(1)(ii).

This report is number 97-019; Docket No. 50-454.

Sincerely,

K. L. Kofton
Station Manager
Byron Nuclear Power Station

KLK/MS/js

Enclosure: Licensee Event Report No. 97-019

cc: A. B. Beach, NRC Region III Administrator
NRC Senior Resident Inspector
INPO Record Center
ComEd Distribution List

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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 MANDATORY 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 (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

LICENSEE EVENT REPORT (LER)

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

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

TITLE (4)

Potential Failure of Westinghouse Fuel Caused by Inadequate Design 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
10	28	97	97	-- 019 --	00	11	24	97	Byron Station, Unit 2	05000455
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	096

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	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)
Ken Kovar, System Engineering Nuclear Group Leader	815-234-5441 X2249

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		12	29	97

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

Westinghouse analysis shows that for high power, high burnup assembly Integral Fuel Burnable Absorber fuel rods, calculated rod internal pressures 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 that Westinghouse used inadequate methods and incorrect assumptions in the fuel rod design code.

Corrective actions are: Westinghouse to review and improve analytical models; gather additional data; and perform plant by plant assessments. Byron Station will evaluate the Byron Unit 2 results scheduled to be received from Westinghouse in mid-December, address this issue in the Byron Unit 1 Cycle 9 safety evaluation and will submit a supplement to this LER.

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

The fuel rod design issues potentially result in a condition outside the design basis reportable per 10CFR73(a)(2)(ii).

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 10-28-97 / 1230

Unit 1 Mode - 1 - Power Operation Rx Power 96.6% RCS [AB] Temperature/Pressure NOT/NOP

Unit 2 Mode - 1 - Power Operation Rx Power 99.9% RCS [AB] Temperature/Pressure NOT/NOP

B. DESCRIPTION OF EVENT:

Background:

Westinghouse Electric Company's fuel performance code is called Performance Analysis and Design (PAD). PAD is a Nuclear Regulatory Commission (NRC) approved methodology that has been in use since the 1970s. Westinghouse uses PAD to demonstrate that fuel performance criteria 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 of the fuel rods and less was being retained in the fuel coating than previously assumed. Because of the few data points available, Westinghouse has since 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 than predicted levels of corrosion being measured in the field on high duty fuel rods. This model was presented to the NRC in December 1996. Westinghouse has been pursuing the incorporation of this new corrosion model into the PAD code and assessing the feedback affects on other fuel performance criteria. During 1996 and 1997, Westinghouse had been applying the revised Zirc-4 corrosion model, without incorporation into the PAD code, to all core designs to confirm less than a steady-state oxide accumulation will be within the design criteria.

With the new corrosion model incorporated, 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.

Current Events:

On 10-28-97 at 1230, Westinghouse reported to Byron Station that based on current modeling for fuel design for a generic limiting analysis, there is a potential to not meet the 10CFR50.46 acceptance criteria for maximum cladding oxidation of 17%, for IFBA fuel which is in the second cycle of operation. Some fuel may potentially exceed the gap reopening design criterion during the second half of the second cycle. If the gap reopening design limit is exceeded, the 17% local oxidation limit following a postulated Loss of Coolant Accident (LOCA) may likewise be exceeded.

B. DESCRIPTION OF EVENT (cont.)

Westinghouse has reported this as a generic issue to the NRC and has supplied a generic Justification for Continued Operation (JCO) for all plants, including Byron Station. In the JCO, Westinghouse has 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 Byron 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 Byron cores is not exceeded. Since performance of the Operability Assessment, Byron Unit 1 has shutdown for a scheduled refueling outage.

On 11-6-97, the Westinghouse Owners Group (WOG) and Westinghouse representatives met with the NRC to provide a technical update on the issue. One of the NRC's top concerns was compliance with the 10CFR50.46 acceptance criteria (17% post-LOCA clad oxidation limit). At the meeting, a screening process was presented to the NRC to address this issue. The issue 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% pre-accident cladding oxidation there is no potential to exceed the 50.46 criteria. The limiting operating plant has been evaluated and will not reach this point for approximately three weeks. Therefore, as of 11-6-97 all operating plants were in compliance with 50.46 criteria. The WOG and Westinghouse committed to providing the NRC, by 11-12-97, with a list of affected plants, and approximate dates when corrosion levels will exceed 12% cladding oxidation during the first quarter of 1998. This list sets the priority for further plant specific evaluations and analyses to demonstrate compliance. Byron Unit 2 is conservatively estimated to potentially exceed 12% cladding oxidation on approximately 12-29-97.

This gap re-opening has the potential to cause the nuclear fuel to be outside the design basis and reportable according to 10CFR73(a)(2)(ii). This issue was reported by Byron Station as a condition potentially outside the design basis of the plant.

C. CAUSE OF EVENT:

Westinghouse initiated a program to obtain additional detailed performance data on its fuel. Many IFBA rods were sent to the Atomic Energy of Canada Limited (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.

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

- 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.

C. CAUSE OF EVENT (cont.)

- 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.

D. SAFETY ANALYSIS:

Gap Re-opening:

Westinghouse has performed a comprehensive safety assessment, which considered the potential for gap reopening. 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 concluded 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 conclusion is consistent with Westinghouse's past fuel operating experience where no evidence of fuel rod failure has been driven by gap re-opening conditions. 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.

17% Cladding Oxidation:

Based on LOCA analysis, Westinghouse has determined that plants with less than 12% pre-accident clad oxidation will not exceed the 10CFR50.46 acceptance criteria of 17%. 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 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-accident and accident 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 is currently being verified, all operating plants, including Byron Unit 2, pass the screening criteria and are therefore in compliance with 10CFR50.46 acceptance criteria.

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 "no gap re-opening" fuel design criteria for IFBA fuel during the second half of the second cycle of operation.

E. CORRECTIVE ACTIONS:

Immediate Actions:

Westinghouse identified a screening criteria that demonstrates that all operating plants are in compliance with 10CFR50.46 acceptance criteria. Therefore, Byron Unit 2 is in compliance.

Continuous 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.

Westinghouse believes that conservatisms in the PAD code 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, Westinghouse will perform 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.

Byron Nuclear Engineering will evaluate the Byron Unit 2 results scheduled to be received from Westinghouse in mid-December. (NTS Number 454-180-97-SCAQ00019-01)

Byron Nuclear Engineering will evaluate this issue in the Byron Unit 1 Cycle 9 reload safety evaluation. (NTS Number 454-180-97-SCAQ00019-02)

Byron Nuclear Engineering will monitor NFS validation of Westinghouse analysis. (NTS Number 454-180-97-SCAQ00019-03)

Byron Corrective Action Department will submit a supplement to this LER to update the Byron Unit 2 results from Westinghouse. (NTS Number 454-180-97-SCAQ00019S1

F. RECURRING EVENTS SEARCH AND ANALYSIS:

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:

There was no component failure.