Commonwealth Edison
Byron Nuclear Station
4450 North German Church Road
Byron, Illinois 61010

November 3, 1994

LTR: BYRON 94-0438
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(a)(2)(ii).

This report is number 94-012; Docket No. 50-454.

Sincerely,

G.K. Schwartz
Station Manager
Byron Nuclear Power Station

GKS/DSK/bl

Enclosure: Licensee Event Report No. 94-012

cc: J. Martin, NRC Region III Administrator
NRC Senior Resident Inspector
INPO Record Center
CECo Distribution List

140014

(9912R/WPF/102494-9)

IE22

SIGNATURE PAGE FOR LICENSE EVENT REPORT

LER Number
454:94-012

Title of Event: Increased Tube Degradation in the Byron Unit 1 Steam Generators

Occurred: 10/06/94/ 1820
Date Time

OSR DISCIPLINES REQUIRED: ABDEG

SG / 10/28/94
SES DATE

Acceptance by Station Review:

B. Glossman 11/1/94
CY Supv ADG

10/28/94
HPSS ABDEG

M Oosmassa / 10-21-94
OE ABCFG Date

/ 10/28/94
SES Date

/ 10-27-94
RAS AG Date

/ 10/27/94
OTHER AG Date

Approved by: / 11/4/94
Station Manager Date

(9912R/WPF/102494-1)

LICENSEE EVENT REPORT (LER)

FACILITY NAME	DOCKET NUMBER	PAGE
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 4	1 OF 0 7

TITLE

INCREASED TUBE DEGRADATION IN THE BYRON UNIT 1 STEAM GENERATORS

EVENT DATE			LER NUMBER			REPORT DATE			OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQ. NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 0	0 6	9 4	9 4	- 0 1 2 -	0 0				None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (CHECK ONE OR MORE OF THE FOLLOWING)			
POWER LEVEL	0		20.402(b)		20.405(e)		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)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER
JAY SMITH, SITE ENGINEERING, EXT. 2604	8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A B		W 1 2 0	N						

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	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).

A steam generator (SG) eddy current inspection was performed in accordance with Technical Specification Surveillance Requirement (TSSR) 4.4.5.0 during the Byron Unit 1 Cycle 6 refuel outage. The results of this inspection classified each of the four SGs as category C-3, as defined in TSSR 4.4.5.2. Category C-3 was declared due to more than 1% of the tubes inspected having degradation that exceeded the repair limit. All defective tubes were removed from service by plugging. The primary mode of the tube degradation was axial oriented Outer Diameter Stress Corrosion Cracking (ODSCC) at the hot leg tube support plates. Packed tube support plate crevices under caustic chemistry conditions with susceptible tube material is considered to be the root cause of the ODSCC.

Preventative measures have been taken to mitigate the initiation of ODSCC. These measures include chemical cleaning, contaminant ingress reduction, and iron transport reduction measures. The occurrence of ODSCC has been experienced in previous outages, but to a lesser extent than the C-3 category. This mode of degradation has been experienced at a number of other plants in the industry. Industry efforts are on-going to determine corrective actions to mitigate ODSCC.

This event is reportable per 10CFR50.73(a)(2)(ii).

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 10/06/94 / 1820

Unit 1 MODE 6 - Core Offload Rx Power 0% RCS [AB]Temperature/Pressure 86°F/0 psig

Unit 2 MODE 1 - Power Op Rx Power 98% RCS [AB] Temperature/Pressure NOT/NOP

B. DESCRIPTION OF EVENT:

During the previous Unit 1 refuel outage (Cycle 5), Byron experienced increased steam generator (SG) tube degradation due to Outer Diameter Stress Corrosion Cracking (ODSCC) at the tube support plate areas. Repair projections for the Cycle 6 refuel inspection indicated that the plugging limit may be exceeded if the current depth based Technical Specification (TS) plugging criteria were used. To reduce the number of repairs due to ODSCC, Byron Station submitted a TS amendment request on August 1, 1994, to implement a voltage based Interim Plugging Criteria (IPC) for the Cycle 6 refuel outage. The original submittal was supplemented on September 7 and September 17, 1994. The proposed IPC would allow flaw indications to remain inservice provided the following criteria are met:

- Dominant mode of degradation is axial ODSCC contained within the confines of the tube support plate.
- Flaw indications are less than or equal to 1.0 volt bobbin coil amplitude.
- Flaw indications between 1.0 and 2.7 volts that are not confirmed by Rotating Pancake Coil examination.
- Flaw indications located outside the areas that are susceptible to collapse during a LOCA+SSE event (wedge locations).

Flaw indications that do not meet the above criteria are considered to exceed the repair criteria and would require repair.

For IPC implementation, enhanced eddy current inspections are required to accurately and reliably determine the voltage amplitude of each indication, since the repair criteria is based on voltage, in lieu of depth. In the August 1, 1994 IPC submittal, Byron committed to perform the inspections in accordance with the IPC inspection guidelines. Following discussions with the NRC concerning the approval schedule of the Byron IPC submittal, it was evident that IPC approval would occur following the inspection and prior to entering Mode 4. Mode 4 entry would be dependent on IPC approval.

On September 20, 1994, steam generator bobbin coil eddy current inspections began pursuant to TSSR 4.4.5.0, including the revision for IPC implementation. The initial sample consisted of 100% of the inservice hot leg and cold leg tubes in each of the four steam generators. Rotating pancake coil inspections were performed on all ODSCC indications greater than 1.0 volt for flaw confirmation in accordance with IPC requirements.

At 1820 on October 6, 1994, at the conclusion of the hot leg eddy current inspection and data evaluation, Byron Station classified the inspection results as category C-3, as specified in TS 4.4.5.2. A C-3 inspection

B. <u>DESCRIPTION OF EVENT:</u> (cont.)

classification is defined as more than 10% of the tubes inspected are degraded (imperfections greater than or equal to 20% throughwall (TW) or more than 1% of the tubes are defective (an imperfection that exceeds the repair limit, 40% TW or 1.0 volt for IPC). Byron determined that the C-3 category requirements were exceeded due to more than 1% of the tubes inspected being defective. The appropriate NRC notification via the ENS phone system was made at 1946 EST on October 6, 1994, pursuant to 10CFR50.72(b)(2)(i). This event is being reported pursuant to 10CFR50.73(a)(2)(ii).

The table below identifies the inspection results and classification of each steam generator:

	SG A	SG B	SG C	SG D	TOTAL
# Inspected (100%)	4392	4279	4335	4459	17465
# Defective Tubes (>/= 40% TW)	6	0	9	2	17
# Defective Tubes (> IPC requirements)	127	167	192	54	540
Total # / (%) Tubes Defective	133 (3.03%)	167 (3.90%)	201 (4.64%)	56 (1.26%)	557 (3.19%)
# /(%) Tubes Degraded	36 (0.82%)	37 (0.86%)	41 (0.95%)	25 (0.56%)	139 (0.80%)
# Tubes Left Inservice through IPC	429	583	572	332	1916
Inspection Category	C-3	C-3	C-3	C-3	C-3

C. <u>CAUSE OF EVENT:</u>

The primary mode of the degradation found is axial oriented ODSCC located at the hot leg tube regions bounded by the tube support plates. The occurrence of ODSCC can be affected by secondary chemistry conditions, crevice conditions, temperature, and tube material properties. Laboratory tests have shown that highly caustic or acidic crevice conditions can promote the initiation and propagation of ODSCC in Inconel 600 tubing with drilled carbon steel support plates.

C. CAUSE OF EVENT: (cont.)

The Byron Unit 1 steam generators are Westinghouse Model D-4s that contain Inconel 600 tubing with carbon steel tube support plates. Review of the secondary chemistry operating history at Byron has indicated that the potential exists for a caustic crevice environment that can be detrimental to the tubes. Secondary side video inspections of the crevice regions, performed during the Cycle 5 refuel outage, verified that the crevices were filled with deposits. As indicated by eddy current, the crevices have become packed over time. This is most likely due to high iron transport to the generators. These packed crevices are believed to be caustic and, therefore, detrimental to the Inconel 600 tubing.

Packed crevices under caustic chemistry conditions with susceptible tube material is considered to be the root cause of the ODSCC and this event.

D. SAFETY ANALYSIS

A safety assessment was performed for the as-found steam generator condition, primarily for ODSCC at tube support plates, as this was the dominant mode of degradation. This assessment involved calculation of the primary-to-secondary leak rate and tube burst probability during a postulated Main Steam Line Break (MSLB). These values were compared to their respective limits as stated in the proposed IPC Technical Specification amendment.

The leak rate and burst probabilities are derived from industry correlations relating eddy current voltage to leak rate and tube burst probabilities. A detailed description of the methods used are contained in the Byron IPC submittal and supplements.

The MSLB leak rate of the as-found SG condition was determined to be 2.3 gpm. This is much less than the site allowable leak rate limit of 12.8 gpm. The site allowable leak rate limit is defined as the maximum SG primary-to-secondary leakage that could occur and meet the offsite dose limit that is based on a small fraction of the 10CFR100 limit. Therefore, from a MSLB leakage perspective, the consequences of this event are acceptable.

From a tube burst perspective, the as-found burst probability was determined to be 1.9×10^{-2}. Although this exceeds the proposed TS tube burst probability limit of 1.0×10^{-2}, the as-found burst probability is less than the 2.5×10^{-2} limit for steam generator burst probability as stated in NUREG-0844. In addition, the following risk assessment demonstrates that the frequency of core damage due to induced rupture of combined single and multiple tubes is less than 10^{-6}, which is the range of many PWR probabilistic risk assessments performed by the industry.

$$(MSLB_{CDF} + FWLB_{CDF}) \times (P_{mit}) \times (P_{rupt}) = CDF$$

$$(1.8\times10^{-3} + 1.8\times10^{-3})/\text{year} \times (10^{-3}) \times (1.9\times10^{-2}) = 6.84\times10^{-8}\ /\text{year}$$

where: $MSLB_{CDF}$ = Main Steam Line Break Frequency (1.8×10^{-3})
$FWLB_{CDF}$ = Feedwater Line Break Frequency (1.8×10^{-3})
P_{mit} = Probability of failure to mitigate events (10^{-3})
P_{rupt} = Total Probability Tube Ruptures (1.9×10^{-2})
CDF = Estimated frequency of core damage

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

D. SAFETY ANALYSIS: (cont.)

As demonstrated above, the estimated incremental risk for the as-found condition would not make Byron an outlier in terms of tube rupture risk from the application of IPC.

Based on the increased repair projections, a reanalysis was performed prior to the Cycle 6 inspection to increase the SG tube plugging limit from 10% to 15%. The total SG tube plugging percentage values of all tubes plugged to date are listed below. These are all less than the revised 15% tube plugging limit. This revised limit was approved by the NRC prior to Mode 4.

	SG A	SG B	SG C	SG D	TOTAL
Total % Plugged	10.0%	11.2%	12.3%	4.8%	9.6%

Reactor coolant system (RCS) flow was evaluated to determine if adequate margin exists. The reanalysis performed to increase the SG tube plugging limit to 15% also provided for additional RCS flow margin. Therefore, adequate RCS flow was determined to exist.

E. CORRECTIVE ACTIONS

Immediate corrective actions included the removal of defective tubes from service. In addition, selected degraded tubes were preventatively plugged due to adverse growth trends. Four tubes were removed from the steam generator in order to perform laboratory testing and analysis. The pulled tube analysis and testing will help gain insight into the mode of degradation and to provide additional data points to the industry leak and burst correlations.

Actions also have been taken to mitigate ODSCC and the corrosive environment in the tube support plate crevices. These actions will ensure that future growth rates and crack morphologies will be within expected bounds.

Steam Generator Chemical Cleaning

Byron chemically cleaned the Unit 1 steam generators during this outage (B1R06). A high temperature process was used to clean the support plate crevice regions. Removal of the crevice deposits will also limit the corrosive environment that is in contact with the tube.

Chemical Additions

- Ethanolamine (ETA) addition to secondary systems will be continued for pH control and to reduce iron transport into the SGs.

- 3-Methoxypropylamine (MPA) addition is being evaluated for pH control and to reduce iron transport into the SGs. NTS #454-180-94-01200-01

E. <u>CORRECTIVE ACTIONS:</u> (cont.)

- Secondary system additives on plant startups will be continued to passivate and reduce the dissolved oxygen levels as well as to maintain a cleanup loop.

<u>SG Crevice pH Improvement</u>

- Steam Generator hideout return studies during shutdowns are performed to assess the impact of operating chemistry on the crevice chemistry. The results of these studies are used to assess the potential formation of caustic crevices.

- The molar ratio control program implemented during Cycle 6 is being re-evaluated for continued use during Cycle 7. This program adjusts the sodium to chloride ratio in the steam generators by adding ammonium chloride to the condensate system. NTS #454-180-94-01200-01

<u>SG Sodium Reduction</u>

- The makeup water regeneration process was modified to reduce the overall impurity input to the secondary side, including sodium.

- The use of a reverse osmosis unit on the makeup water system to reduce impurities is being evaluated. NTS #454-180-94-01200-01

- Eddy Current inspection was performed on 100% of the tubes in all 4 of the main condenser waterboxes during the Cycle 5 refuel outage. Eddy current inspection was also performed on 100% of the tubes in one waterbox during the Cycle 6 refuel outage. These inspections were performed to identify and repair tubes that may cause leakage and contaminant ingress to the steam generators.

<u>SG Electrochemical Potential (ECP) Reduction</u>

Byron is using elevated hydrazine in the secondary system for maintaining reducing conditions in the steam generators and passivation of piping systems and components. Use of alternate additives are currently being evaluated. NTS #454-180-94-01200-01

Currently, Byron is pursuing several methods to further enhance the steam generator corrosion control program, in conjunction with the Corporate Chemistry Department. These methods are as follows:

- The addition of other amines with either ETA or in lieu of ETA is being evaluated to optimize the pH control and minimize iron transport into the steam generators. NTS #454-180-94-01200-01

- Chemical controls are being evaluated to improve iron transport out of the steam generators. The goal is to increase the efficiency of iron removal via the steam generator blowdown system. NTS #454-180-94-01200-01

FACILITY NAME	DOCKET NUMBER	LER NUMBER YEAR	SEQ. NUMBER	REVISION	PAGE
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 4	9 4	- 0 1 2	- 0 0	0 7 OF 0 7

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

F. RECURRING EVENTS SEARCH AND ANALYSIS

This is the first occurence of a Byron Unit 1 or Unit 2 steam generator being classified as inspection category C-3. Historically, Byron Unit 1 has experienced ODSCC at the tube support plates, but not to the extent of a C-3 classification.

Braidwood Unit 1 experienced a similar event during their Cycle 4 (Spring 1994) inspection when three of the four steam generators identified with a C-3 classification due to an increase of repairs from ODSCC at tube support plates. (LER #456:94-007)

ODSCC at tube support plates has been experienced at a number plants throughout the industry. Industry efforts are on-going to understand and correct this mode of degradation. ComEd is actively involved in these efforts.

G. COMPONENT FAILURE DATA

Manufacturer	Nomenclature	Model Number	MFG Part Number
Westinghouse	Steam Generator	D-4	n/a