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

November 3, 1989

Ltr: BYRON 89-1067

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)(v).

This report is number 89-009-00; Docket No. 50-454.

Sincerely,

for R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/bb (0459R/0059R)

Enclosure: Licensee Event Report No. 89-009-00

cc: A. Bert Davis, NRC Region III Administrator
W. Kropp, NRC Senior Resident Inspector
INPO Record Center
CECo Distribution List

IE22
1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5	1 of 0 9

Title (4) INADEQUATE INCORPORATION OF STEAM GENERATOR BLOWDOWN ISOLATION REQUIREMENTS AS ASSUMED IN CERTAIN ACCIDENT ANALYSIS CAUSED BY A PRESERVICE DESIGN IMPLEMENTATION DEFICIENCY

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 Names	Docket Number(s)
1 0	0 5	8 9	8 9	0 0 9	0 0	1 1	0 3	8 9	BYRON UNIT 2	0 5 0 0 0 4 5 5
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	X 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)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 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 (12)

Name	TELEPHONE NUMBER (AREA CODE)
Gary Stauffer, Assistant Technical Staff Supervisor Ext. 2274	8 1 5 2 3 4 - 5 4 4 1

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)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

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

During review of erosion/corrosion concerns for the Steam Generator blowdown (SD) lines, conflicting information regarding signals that initiate automatic isolation of the SD lines was discovered by Byron site personnel. The SD isolation was confirmed to occur on a Containment Phase A signal (generated on receipt of any Safety Injection Signal (SIS)) and an Auxiliary Building high temperature signal (high energy line break consideration). However, the Updated Final Safety Analysis Report (UFSAR) indicated that automatic SD isolation would also occur on other signals.

Discussions with Westinghouse indicated that SD isolation on a SIS and 2 of 4 Steam Generator (SG) level low-low signal were assumed in certain accident analyses. The effect of not isolating SD when required is to reduce the Auxiliary Feedwater (AF) flow provided to the SG for cooling.

As interim corrective action SD was isolated at 12:00 on 10/05/89 until temporary procedure changes were implemented that required manual isolation of blowdown whenever a reactor trip occurs to replicate the UFSAR accident analysis.

An engineering evaluation was provided to the station on 10/25/89 which demonstrated that the assumptions of the accident analyses have satisfactorily been met for past and interim operations in conjunction with the revised operating procedures until a permanent modification can be implemented.

The proximate cause of the event was a preservice design implementation deficiency. The SD system functional requirements identified several signals to provide SD isolation, however isolation on SG level low-low was not physically incorporated for unknown reasons.

This event is reportable per 10CFR50.73(a)(2)(v). There have been no previous similar events.

(0459R/0059R)

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 10/05/89 / 1200

Unit 1 MODE 1 - Power Operation Rx Power 100% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 1 - Power Operation Rx Power 98% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

During a review of erosion/corrosion concerns for the Steam Generator blowdown (SD)[WI] lines, conflicting information regarding signals that initiate automatic isolation of the SD lines was discovered by Byron site personnel. UFSAR Figure 7.2-1 sheet 15 of 18 (Westinghouse drawing 108D685) indicates the SD and sample line valves (1/2SD002A through H and 1/2SD005A through D respectively) close on the following signals: (1) local manual start of the Auxiliary Feedwater (AF) [BA] Pumps (2) Control Room Manual start of the AF Pumps (3) Safety Injection Signal (SIS) (4) Loss of Power signal (2 of 4 Reactor Coolant Pump bus undervoltage) and (5) 2 of 4 Steam Generator (SG) [AB] level low-low. However, the UFSAR text does not discuss all of these isolation signals. Section 9.3.2.2.2 indicates Steam Generator blowdown process and sample containment isolation valves are automatically closed on high containment pressure (containment pressure high-1 would generate a SIS). Section 10.4.8 states blowdown lines from each steam generator have two air operated valves capable of auto closure. Also Table 15.1-2 identifies equipment required following a rupture of a main steam line which includes the SD isolation valves auto closure feature. Review of the Byron Station schematic diagrams for the SD isolation and sample valves was completed and indicated isolation is only provided by (1) Containment Phase A isolation signal (Phase A is generated by any SIS) and (2) Auxiliary Building high temperature (High Energy Line Break (HELB) consideration).

On 10/02/89 this discrepancy was discussed with Westinghouse. Westinghouse indicated that SD and sample valve isolation on only a SIS and 2 of 4 steam generator level low-low signal were assumed in certain accident analyses. The AF system provides water to the Steam Generators to remove residual and decay heat from the Reactor Coolant System (RCS). The effect of not isolating the SD and sample lines is to reduce the net amount of AF water delivered to the SG. For those accident scenarios relying on AF, this can impact the Departure from Nucleate Boiling Ratio (DNBR) and result in system overpressurization. Westinghouse also identified a series of letters written in 1980 regarding the potential for reduced secondary cooling with SD isolation valve failures. The Westinghouse to CECo letters dated 02/20/80, 04/11/80 and 05/09/80 were retrieved from Westinghouse files on 10/03/89. The letters recommended that the design of the AF and SD Systems be evaluated to ensure that the net flow to the intact steam generators is not less than the AF requirements as stated in the SAR analysis. The letters also stated that if the analysis performed showed that the licensing basis was marginally met, or not met, consideration should be given to making plant modifications such as adding an additional SD isolation valve or adding HELB restraints. A Sargent & Lundy to CECo letter dated 05/14/80 addressed this issue and determined that no new valves or restraints were required. These Westinghouse letters did not specifically discuss SD isolation on a SG level low-low signal and the Sargent & Lundy letter did not identify any concerns with not having SD isolation on SG level low-low.

The preliminary station and engineering review completed on 10/05/89 could not conclude that AF actuation without SD isolation had been adequately resolved in the past. Therefore as a conservative action, valves 1/2SD002A through H and sample valves 1/2SD005A through D were closed at 12:00 on 10/05/89, thereby securing SG blowdown on both units. The appropriate NRC notification via the ENS phone system was made at 1253 pursuant to 10CFR50.72(b)(2)(iii). Plant operation with the SD valves isolated could not continue for any extended period of time without adverse consequences on the SG secondary chemistry which would ultimately require a unit shutdown. Temporary procedure changes were implemented for 1/2BEP 0 (Reactor Trip or Safety Injection), 1/2 BCA 0.0 (Loss of all AC Power), 1/2BFR S.1 (Response to Nuclear Power Generation/ATWS) and 1/2 BFR H.1 (Response to Loss of Secondary Heat Sink), requiring operators to close the steam generator blowdown and sample isolation valves on any reactor trip signal. Procedures 1/2BFR H.5 (Response to Steam Generator Low Level) already had contained a step requiring SD isolation. Since the temporary procedure changes replicated the assumptions in the accident analysis, On-site Review 89-227 dated 10/05/89 concluded that it was acceptable to re-establish SG blowdown until CECo engineering and Westinghouse could conduct a more detailed review of plant construction documentation and of additional accident analyses to verify that AF actuation without SD isolation had already been analyzed or considered if required. A daily order was issued to operations personnel on 10/05/89 notifying the operators of the temporary procedure changes and training was also conducted for shift personnel.

On 10/25/89 the Station received the engineering evaluation of AF actuation without automatic SD isolation performed by CECo Engineering and Westinghouse. The evaluation concluded that even though the accident analyses assumed SD isolation on a SG level low-low signal, the AF system would still be capable of performing its intended function to mitigate the consequences of the affected LOCA and non-LOCA transients. It also concluded that interim operation was acceptable with the temporary procedure revisions currently existing based on the reanalysis done until a permanent modification design and implementation could be completed. An On-Site Review 89-245 was performed which accepted the results of this engineering evaluation.

There were no systems or components inoperable at the beginning of the event that contributed to the event. Operator actions did not impact this event. The event is reportable per 10CFR50.73(a)(2)(v).

C. CAUSE OF EVENT:

The proximate cause of the event was a preservice design implementation deficiency. The SD system functional requirements identified several signals to provide SD isolation, however isolation on SG level low-low was not physically incorporated.

The SD System functional requirements identified other isolation signals in addition to the SIS and SG level low-low signals that Westinghouse currently states are the SD isolation signals assumed in the accident analysis. The Steam Systems Design Manual SG-689 issued in 1979 indicates the blowdown and sample containment isolation valves must be designed to close automatically when AF receives an automatic actuation signal. There are several AF actuation signals that do not provide SD isolation (2 of 4 RCP bus undervoltage and ESF bus undervoltage). UFSAR Figure 7.2-1 sheet 15 of 18 (prepared from Westinghouse logic drawing 108D685) indicates that Westinghouse provided the required blowdown and sample line valve isolation signals, but other vendors provided the physical wiring to install the isolation signal. Again this figure has several signals that in fact do not provide SD isolation, including manual start of the AF pumps. The series of letters issued in 1980 regarding the potential for reduced secondary cooling with SD isolation valve failures did not specifically address SD isolation on SG level low-low and the design concern was not recognized at this time.

The SD isolation valves, 1/2SD002A through H and 1/2SD005A through D, are included in Technical Specification 3/4.6.3 for containment isolation valves. The AF system is discussed in Technical Specifications 3/4.3.2 and 3/4.7.1.2 and the SD isolation valves are not addressed in these specifications. Therefore the impact on AF operability from the lack of SD isolation on a SG level low-low signal would not be apparent from the Technical Specifications.

Since the original design discussions occurred approximately 10 years ago, it is difficult to determine the root cause of the preservice design implementation deficiency. As noted, there are several SD isolation signals discussed as part of the original design that were not incorporated into the final plant construction and are determined not to be required since they are not assumptions in the accident analyses. The SG level low-low SD isolation signal may have been addressed in the original design, but there is insufficient documentation to arrive at this conclusion. No further investigations are planned.

D. SAFETY ANALYSIS:

The design function of the AF system is to provide adequate cooling water to the steam generators to remove residual and decay heat from the RCS when the normal feedwater system is unavailable. The AF system allows the RCS to be cooled to a temperature where the residual heat removal system can be placed in operation. The effect of not isolating the steam generator blowdown and sample lines is to reduce the net amount of AF water delivered to the SGs. For those accident scenarios relying on AF, failure to isolate SD and the sample valves can impact the DNBR and system overpressurization. The following is a discussion of the effects on the non-LOCA and LOCA accident analysis.

NON-LOCA IMPACT - The impact of not automatically isolating the steam generator blowdown and sample lines affects those transients in which auxiliary feedwater (AF) is relied upon to remove long term core decay heat. These events include the Loss of Normal Feedwater event (UFSAR 15.2.7), the Loss of Non emergency AC Power to the Plant Auxiliaries (UFSAR 15.2.6), and the Feedwater System Pipe Break (UFSAR 15.2.8). The remaining Chapter 15 events are not impacted by a reduction in AF delivery. In order to quantify the effects of the reduction in AF flow for the affected accidents, the cases presented in the UFSAR were reanalyzed as follows.

LOSS OF NORMAL FEEDWATER/LOSS OF NON-EMERGENCY AC POWER - The Loss of Normal Feedwater/Loss of AC Power events are ANS Condition II transients. Thus, the acceptance criteria are based on DNBR and system overpressurization. In addition, the plant must be capable of returning to operation after corrective action and the event can not generate a more severe transient. Since these events are bounded by other transients with respect to potential fuel damage, DNBR is not a concern. The Westinghouse acceptance criterion, therefore, is based on not permitting the pressurizer to go water solid. This ensures that the RCS does not overpressurize and that primary effluent does not spill to containment.

In order to generate margin to offset the AF penalty of not immediately isolating SD, better-estimate analyses were performed to justify previous operation. The assumptions that went into these analyses are detailed below. Note that the single failure assumption of one of the two AF pumps failing to start was not changed. If the single failure assumption is removed, then the flow provided by the second pump more than offsets that lost via the blowdown lines and new analyses would not be required.

a. The initial power level was based on 100% of the currently licensed NSSS rating (3425 MWt) instead of the UFSAR assumption of 102% of the ESF rating (3579 MWt). This is a reduction of 225 MWt and is based on actual plant operating conditions.

b. The remainder of the uncertainties on initial conditions were removed. Nominal values for RCS temperature, feedwater temperature, and pressurizer pressure were used.

c. A nominal value of 14 MWt for reactor coolant pump (RCP) heat was used rather than a maximum value of 20 MWt.

d. The actual plant value of 27 ft^3 for the AF piping purge volume was assumed instead of the conservative UFSAR value of 50 ft^3.

e. The SD lines were assumed to remove 90 gpm per steam generator. This value bounds all four Byron and Braidwood units.

f. The operators were assumed to manually isolate the blowdown lines from the control room at 10 minutes.

g. Based on actual AF performance calculations, 160 gpm was assumed to be delivered to each steam generator once the isolation valves are closed. This corresponds to 640 gpm supplied to all 4 steam generators. Prior to isolation, 70 gpm was assumed to be delivered to each steam generator.

h. Core residual heat generation was based on the 1979 version of ANS 5.1 including a 2-sigma uncertainty. ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates.

Two sets of analyses were performed to encompass all four units (Byron and Braidwood Units 1 and 2). One analysis assumed nominal conditions based on an RCS average temperature of 588.4°F. The other analysis was based on an RCS average temperature of 569.1°F. The two analyses were necessary because several of the units operate under the Thot Reduction Program but Braidwood Unit 2 does not. This program is detailed in WCAP-11386.

The results of these analyses demonstrated that all the UFSAR acceptance criteria would still be satisfied given the assumptions identified above. The nominal temperature cases resulted in peak pressurizer water volumes of 1548 ft^3 and 1549 ft^3 for the Loss of Normal Feedwater and Loss of AC Power events, respectively. For the reduced temperature cases, the peak pressurizer water volumes were 1504 ft^3 and 1619 ft^3, respectively. The pressurizer PORVs and main steam safety valves (MSSVs) limit peak primary and secondary pressures to less than 110% of design. The pressurizer safety valves would have prevented overpressurization had the PORVs been unavailable.

Therefore, based on these better-estimate analyses, the UFSAR criteria were satisfied and, subsequently, the conclusions presented in the UFSAR remain valid for past operation.

FEEDWATER SYSTEM PIPE BREAK - The Feedline Break event also relies on the delivery of AF to remove residual and decay heat from the RCS. In order to quantify the effects of the reduction in AF on this event, the two UFSAR cases (both with and without offsite power available) were reanalyzed.

The Feedline Break event is an ANS Condition IV transient. The applicable acceptance criterion is to demonstrate that the core remains intact and in a coolable geometry. The more restrictive Westinghouse criterion is to show that the RCS remains subcooled, thus ensuring that bulk boiling in the hot legs does not occur.

Consistent with the analyses discussed above, better-estimate assumptions were used in this analysis to justify previous operation. Again, if the single failure assumption is removed, then the flow provided by the second pump more than offsets that lost via the blowdown lines and a new analyses would not be required.

a. The initial power level was based on 100% of the ESF rating (3579 MWt) instead of 102%. This is a reduction of 72 MWt and reflects nominal uprated plant performance.

b. The remainder of the uncertainties on initial conditions were removed. Nominal values for RCS temperature, feedwater temperature, and pressurizer pressure were used.

c. A nominal value of 14 MWt for reactor coolant pump (RCP) heat was used rather than a maximum value of 20 MWt.

d. The actual plant value of 27 ft^3 for the AF piping purge volume was assumed instead of the conservative UFSAR value of 50 ft^3.

e. The SD lines were assumed to remove 90 gpm per steam generator. This value bounds all four Byron and Braidwood units.

f. The blowdown lines were assumed to be isolated by a SIS generated by the Low Steamline Pressure function and is supported by existing plant logic (containment phase "a" isolation on any SIS signal). This signal is actuated about 6 minutes into the transient.

g. Based on actual AF performance calculations provided by the utility, 160 gpm was assumed to be delivered to each steam generator once the isolation valves are closed. This corresponds to 480 gpm supplied to the 3 intact steam generators. Prior to isolation, 70 gpm was assumed to be delivered to each steam generator.

h. Core residual heat generation was based on the 1979 version of ANS 5.1 including a 2-sigma uncertainty. ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates.

Only one analysis was performed for Feedline Break to address all four units because WCAP-11386 concluded that nominal temperature conditions (i.e., based on a RCS average temperature of 588.4°F) are limiting.

The results of this analysis demonstrated that the UFSAR acceptance criterion would still be satisfied given the assumptions identified above.

The results showed that the minimum subcooling margin for both the with and without offsite power cases was 33°F. The pressurizer safety valves and MSSVs limit peak primary and secondary pressures to less than 110% of design. Therefore, based on this better-estimate analysis, the UFSAR criteria is satisfied and, subsequently, the conclusions presented in the UFSAR remain valid for previous operation.

STEAMLINE BREAK MASS & ENERGY RELEASES - The reduction in AF has the potential to impact the Steamline Break Mass & Energy Releases calculated both inside containment for containment integrity (UFSAR 6.2.1.4) and outside containment for equipment qualification (WCAP-10961-P).

For the mass and energy releases inside containment, primary protection is provided by either a Containment Pressure High-1 or a Low Steamline Pressure signal. Both of these functions generate a SIS as well as reactor trip and AF actuation. Thus, the SD isolation valves will be closed and no AF will be lost. Therefore, this scenario does not impact the mass and energy releases inside containment.

For the mass and energy releases outside containment, the program detailed in WCAP-10961-P was developed to identify what equipment modifications, on a forward fit basis, were necessary to ensure proper operation of safety-related equipment in the auxiliary buildings. Since these equipment modifications only address future operation, it is inappropriate to consider this scenario against the generation of the mass and energy releases for past operation. The auxiliary feedwater performance currently assumed in this program for Byron/Braidwood will be preserved for both interim and permanent operation in the future.

LOCA IMPACT - The calculated consequences for both the large and small break LOCA analyses appearing in the UFSAR assume isolation of the SD lines. This isolation will occur on a SIS as a result of pressurizer low pressure. Since the isolation function is provided for a SIS the UFSAR large and small break LOCA analyses are not affected. Further, the Thot reduction analyses (WCAP-11387 Rev.1) and the VANTAGE 5 fuel analyses are not affected.

In addition, the LOCA analyses or licensing positions that are not affected by assumptions for Steam Generator Blowdown line isolation include 1) LOCA hydraulic forcing function analyses used to develop LOCA loads on the reactor internals and RCS loop piping, 2) Post-LOCA boron concentration required to keep the reactor subcritical and 3) Switchover of the ECCS to hot leg recirculation required to prevent the potential for boron precipitation.

CONCLUSION - The effects of the scenarios identified above, involving the lack of automatic isolation of the steam generator blowdown and sample lines on an SG level low-low signal (AF actuation signal), have been evaluated against the non-LOCA and LOCA accident analyses. This evaluation has concluded that while the plants did not have SD isolation on SG level low-low assumed in the accident analyses, at no time were the plants in a condition that posed significant risk to public health and safety. These better-estimate Loss of Normal Feedwater, Loss of AC Power, and Feedline Break analyses conclude that with the assumptions itemized above, the UFSAR acceptance criteria are satisfied and the conclusions of the UFSAR are still valid. All other UFSAR Chapter 15 transients are unaffected by this potential scenario.

The better-estimate assumptions used in the above analyses are permissible in addressing previous operation, and for future operation provided that operator action to isolate the SG blowdown lines is accomplished in 10 minutes following a reactor trip and this requirement is included in the plant operating procedures. Prior to the temporary procedure changes, the requirement to isolate SD did exist in the functional restoration procedure 1/2BFR H.5 (Loss of Heat Sink). In addition many station emergency procedures require verification that the steam generator level is greater than 4% of narrow range. If not, the operator is instructed to maintain AF flow greater than 500 gpm until narrow range level is greater than 4% in at least one steam generator. This allows the operator to evaluate the situation and increase AF flow to greater than 500 gpm if level is not being restored. It is reasonable to assume the operator would also have recognized the need to isolate SD if there was any difficulty in restoring SG level. Under the worst case condition of sustained inadequate AF flow, the emergency procedures provide for either the establishment of feed to the SGs from the normal feedwater system or cooldown and depressurization of the RCS to a point where the Residual Heat Removal System can be placed in service using redundant ECCS components.

It is worthwhile to reiterate that had the limiting single failure assumption of one AF pump been removed, no analyses would have been required since a second pump more than makes up for the lost AF flow. In addition, a feedwater system pipe break that assumes SD isolation on SG level low-low has never occurred at Byron Station. However, there have been Loss of Normal Feedwater and Loss of Non-Emergency AC Power events but review of Station documentation regarding these events indicates no adverse impact has occurred because of lack of SD isolation on SG level low-low. The Byron and Braidwood stations are justified to continue to operate, based on the discussions presented above in conjunction with the revised operating procedures indicating a manual isolation of the SG blowdown and sample lines on any reactor trip signal. Closing the SD and sample isolation valves on any reactor trip signal is more conservative than the assumptions in the accident analysis which assume isolation only on a SIS or Steam Generator level low-low signal. Also, a permanent modification will be completed on a prudent schedule during the next outage of sufficient duration allowing for engineering design, material procurement and outage scheduling.

E. CORRECTIVE ACTIONS:

When the preliminary review could not conclude that AF actuation without automatic SD isolation on SG level level low-low was an acceptable design, the SG blowdown valves 1/2SD002A through H and the sample isolation valves 1/2 SD005A through D were closed. Temporary procedure changes were implemented for 1/2 BEP 0, 1/2 BCA 0.0, 1/2 BFR S.1 and 1/2 BFR H.1 requiring operators to close the steam generator blowdown and sample isolation valves on any reactor trip signal. These procedure changes replicated the assumptions in the accident analysis for automatic SD isolation on SG level low-low and Station On-Site Review (OSR 89-227 dated 10/05/89) determined it was acceptable to re-establish SG blowdown until CECo engineering and Westinghouse conducted a more detailed review of plant construction documentation and of additional accident analyses to verify that AF actuation without SD isolation had already been analyzed or considered if required.

On 10/25/89, the Station received the engineering evaluation of AF actuation without automatic SD isolation performed by CECo Engineering and Westinghouse. A station On-site Review (OSR 89-245 dated 10/31/89) determined that interim operation was acceptable with the temporary procedure revisions currently existing, based on the reanalysis performed, until a permanent modification can be implemented. The permanent modification will be completed on a prudent schedule during the next outage of sufficient duration allowing for engineering design, material procurement and outage scheduling. The tentative schedule for Byron Unit 1 is the fourth refueling outage anticipated to commence 09/17/91 and the second refueling outage for Byron Unit 2 anticipated to commence 09/09/90. Action Item Records (AIR) 454-225-89-29200 and 455-225-89-29300 will track the design development and installation of the modification. UFSAR Figure 7.2-1 will be revised to reflect only the signals required to provide SD isolation. These AIRs will also track any permanent procedure and other document changes required.

F. PREVIOUS OCCURRENCES:

A historical review of Byron's LERs has identified no known previous similar events.

G. COMPONENT FAILURE DATA:

Not applicable.