U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
REGION IV

Report No. 99900404/80-01 Program No. 51100

Company: Westinghouse Electric Corporation
Nuclear Technology Division
Post Office Box 355
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Inspection Conducted: January 21-24, 1980

Inspector: Donald Gene Anderson 6/6/80
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C. J. Hale, Chief Date
Program Evaluation Section
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Inspection Summary

Special inspection on January 21-24, 1980 (Docket No. 99900404/80-01)

Areas Inspected: Review of procedures and controls adopted by Westinghouse to implement the requirements of 10 CFR Part 21 including follow-up and review of 10 CFR Part 21 and 10 CFR 50.55(e) licensee reports for evaluation and reporting per 10 CFR part 21. The inspection involved fifty-two (52) hours onsite by two (2) NRC inspectors.

Results: No violations or deviations were identified. One unresolved item was identified.

Unresolved Item: Sufficient documentation was not examined to verify that all customers receiving defective Raytheon RC 747D integrated circuit chips had been notified. (See Details Section II, paragraph F.4.)

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DETAILS SECTION I

(Prepared by D. G. Anderson)

A. Persons Contacted

G. Butterworth, Senior Engineer
C. L. Gottshall, Lead Engineer
*E. J. Hampton, Manager, Product Assurance Systems
M. H. Judkis, Project Manager
N. W. Kish, Senior Engineer
G. E. Lang, Senior Engineer
*P. T. McManus, Senior Quality Engineer
D. H. Rawlins, Manager, Standards and Electrical Systems Evaluation
*R. A. Wiesemann, Manager, Regulatory and Legislative Affairs

*Indicates attendance at the exit meeting.

B. Follow-up on Various Reports by Westinghouse

1. Objectives

The objectives of this area of inspection were to follow-up on reports which had been initiated by Westinghouse and either reported directly by Westinghouse or by utilities when notified by Westinghouse. In reviewing these reports, the inspector assured that the following objectives were accomplished:

a. Determination of how the items were identified.

b. That followup actions were conducted under the requirements and procedures of the Westinghouse quality assurance program.

c. Determination of the status of corrective action and preventive action to assure that the items are satisfactorily resolved.

d. Determination of generic effects on other Westinghouse plants and notification of affected utilities.

e. Determination of the accuracy and timeliness of reporting to the NRC.

2. Method of Accomplishment

The inspector reviewed the following documentation related to the six reportable items in the paragraphs below to assure that the above noted objectives were accomplished on each item reported.

a. Policies and procedures implemented by Westinghouse/NTD to meet the reporting requirements of 10 CFR 50.55(e) and 10 CFR 21 are described in the following documents:

(1) Westinghouse Topical Report WCAP 8370.

(2) Westinghouse Product Assurance Manual.

(3) WRD-OPR-19.0, Identification and Reporting of Substantial Safety Hazards, Significant Deficiencies, and Unreviewed Safety Questions, 7/19/79.

b. Safety Review Committee Manual, January 3, 1978, which includes the following:

(1) Tab 10, Committee Procedural Rules, which identifies meeting frequency, quorum, voting, minutes, and notification to the NRC under 10 CFR 50.55(e) and 10 CFR 21.

(2) Tab 11, Guidelines for Referral of potential items to the Safety Committee.

(3) Current membership list of the Safety Review Committee.

(4) Tab 14, potential items status list. This list contains 153 items reviewed by the Safety Review Committee since 1974.

3. Environmental Qualification of Non-Safety Grade Systems

This item is related to the impact on the protective functions performed by safety related equipment caused by an adverse environment resulting from a high energy line break inside or outside of containment. Westinghouse has identified four systems which could be affected.

a. The inspector determined that the following utilities (plants) reported this item as a 10 CFR 50.55(e) after meeting with Westinghouse representatives in Monroeville, Pennsylvania on September 6, 1979 SNUPPS, Salem 1, McGuire, Marble Hill 1 and 2, North Anna, and Surry.

It is possible that this item may be generic to all pressurized water reactors (PWRs).

b. The Safety Review Committee completed a review of this potential item on August 28, 1979, and this evaluation is documented in ID-79-144.

The NRC issued IE Information Notice No. 79-22 on September 14, 1979, and then the NRC staff met with Westinghouse at Monroeville on September 19 and reviewed this item.

c. Corrective action on this item is identified in the Safety Review Committee Meeting minutes of September 24, 1979 (NS-RAW-139). The Committee identified four systems (items) which were impacted by a high energy line break inside or outside of containment and determined the following:

(1) The four items are not substantial safety hazards.

(2) The four items are unreviewed safety questions.

(3) The four items are potential significant deficiencies for plants under construction.

d. The inspector reviewed the following documents during this inspection which related to this item:

(1)	NS-RPA-I-722	8/15/78	PRESSURIZER POWER OPERATED VALVES CONTROL SYSTEM
(2)	NS-RPA-I-748	8/25/78	LEVEL REFERENCE LEG WATER DENSITY
(3)	NS-RPA-I-764	9/7/78	CONTROL SYSTEMS ENVIRONMENTAL REQUIREMENTS
(4)	NS-RPA-I-787	9/21/78	CONTROL SYSTEM OPERATION
(5)	NS-RPA-I-820	10/31/78	CONTROL SYSTEM ENVIRONMENTAL QUALIFICATIONS
(6)	NS-RPA-I-837	11/20/78	STEAM GENERATOR REFERENCE LEG DENSITY EFFECTS
(7)	NS-RPA-I-864	12/15/78	CONTROL SYSTEM INTERACTION - ROD CONTROL SYSTEM
(8)	NS-RPA-I-865	1/22/79	GROUNDRULES FOR ACCIDENT ANALYSIS
(9)	NS-RPA-I-922	2/8/79	DESIGN BASIS FOR INSTRUMENT ENVIRONMENTAL ACCURACY REQUIREMENTS
(10)	NS-RPA-I-927	2/20/79	CONTROL SYSTEM INTERACTION - ROD CONTROL SYSTEM
(11)	NS-RPA-I-928	2/21/79	CONTROL SYSTEM INTERACTION - STEAM GENERATOR POWER OPERATED RELIEF VALVE CONTROL SYSTEM

(12)	NS-RPA-I-926	3/21/79	CONTROL SYSTEM INTERACTION - PRESSURE CONTROL SYSTEM
(13)	NS-RPA-I-963	4/16/79	CONTROL SYSTEM INTERACTION - MAIN FEEDWATER CONTROL SYSTEM
(14)	AE-TCE-514	5/15/79	ENVIRONMENTAL QUALIFICATION - STEERING COMMITTEE CONTROL SYSTEMS INTERACTION
(15)	NS-RJS-79-043	7/10/79	CONTROL AND PROTECTION INTERACTIONS

e. Follow-up Item

The inspector noted that as a result of a Safety Review Committee evaluation (ID-79-144) of four transients resulting from a high energy line break inside or outside of containment, that it is not apparent why this item was not reported under the requirements of 10 CFR Part 21 as a substatial safety hazard. In particular, the preamble to Part 21 defines (change 5) substantial safety hazard as moderate exposure to, or release of licensed material. NUREG-0302 (Rev 1) on page 21.3(k)-1 further defines moderate exposure as "Exposure in excess of 25 rems, whole body (10 CFR 20.403)". Westinghouse has failed to evaluate the effects of these transients on whole body exposures to individuals inside and outside of containment. This item is similar to the discussion in Details Section, paragraph B.4.a in Report No. 99900404/79-04.

4. Main Feedwater Water Hammer Analysis

This item relates to a reanalysis performed by Westinghouse which indicates that in the event of a possible water hammer in the main feedwater line between a check valve and the steam generator, pressures of several thousand psi could result.

a. This item was reported by Sequoyah 1 and 2 on January 15, 1979, under the requirements of 10 CFR 50.55(e) and indicated that this concern could possibly be generic to all PWRs.

Westinghouse had previously evaluated this item and determined it to be not reportable in 1977 (PI-77-14, PI-77-44) and again in 1978 (ID-78-135, ID-78-138).

b. Westinghouse issued Technical Bulletin NSD-TB-79-8, Water Hammer in Steam Generator/Feedwater Lines, Feedwater Piping to Steam Generator, November 26, 1979. This bulletin reported that this item could possibly be generic to all PWRs and the following utilities (plants) were notified;

Catawba 1 and 2, McGuire 1 and 2, Sequoyah 1 and 2, Watts Bar 1 and 2, Virgil C. Summer, and Shearon Harris.

c. The inspector reviewed the following documents during this inspection which related to this item:

(1) BOPSD-M-1096, Transmittal of Water Hammer Information for Preheat Steam Generator (PI-78-44) June 23, 1978.

(2) LP-1010, Safety Review Committee meeting minutes (ID-78-135) June 28, 1978.

(3) NS-RAW-075, Feedwater Line Break Closeout (PI-77-17) October 25, 1978.

(4) NS-RAW-075, Water Reactor Division/Safety Review Committee meeting-Feedwater Break Transient Analysis (ID-78-138) February 13, 1979.

5. Dropped Rod Accident Analysis

This item was identified during a Westinghouse review of the rod drop analysis. It has been determined that the dropped rod accident could result in DNBR as reported in the SAR being non-conservative, that errors involved in the high flux rate trip circuitry could permit the actual rate being attained and the trip not occurring, and that multiple dropped rods may not actually result in a reactor trip.

a. The inspector verified that this item was identified and evaluated by Westinghouse (NS-RPAI-995, Safety Review Committee Meeting-ID-79-140, on March 27, 1979) and subsequently reported to the NRC under the requirements of 10 CFR 50.59 as an unreviewed safety question (NS-TMA-2063, dated March 30, 1979). On November 15, 1979, Westinghouse reported this item to the NRC as an unreviewed safety question and a potential significant deficiency. This report contains generic applicability to all Westinghouse operating and non-operating plants as appropriate (NS-TMA-2162).

Westinghouse notified all affected utilities on November 15, 1979 (ALA-TRP-1675).

The Safety Review Committee meeting minutes of November 14, 1979 addresses short term corrective action and long term preventive action (addition of a safety grade rod block).

Westinghouse met with the NRC in Bethesda on November 19, 1979, to resolve this issue. Document NS-TMA-2167 (Rod Drop Analysis, November 28, 1979) identifies the scope and results of that meeting.

b. The inspector reviewed the following documents related to this item:

(1) SBN-109, Seabrook Station 10 CFR 50.55(e) Interim Report on Rod Analysis, December 13, 1979.

(2) 10 CFR 50.55(e) "Significant Deficiency" Dropped Rod Event No. 2 Unit Salem Generating Station, December 3, 1979.

(3) FNP-79-0415, Licensee Event Report-Single Dropped Rod Analysis, Farley Nuclear Plant, March 30, 1979.

(4) SLNRC 79-22, 10 CFR 50.55(e) Report-Rod Drop Analysis-SNUPPS, December 14, 1979.

6. Safety Injection System Design Inadequacy

This item was identified by a licensee during containment venting operations. In particular, the venting occurred with the containment high particulate radiation monitor isolation signal to the purge and pressure-vacuum relief valves overridden. After evaluation of this practice, the licensee determined that the reset of the particulate alarm also bypasses the containment isolation signal to the purge valves and the purge valves would not have automatically closed in the event of an emergency core cooling system (ECCS) safety injection signal.

a. This item was reported by Salem 1 as a prompt LER and by Salem 2 as a 10 CFR 50.55(e) construction deficiency on September 8, 1978.

IE Circular 78-19, Manual Override (Bypass) of the Safety System Actuation Signals, was issued by the NRC on December 29, 1978, and was based upon reporting by Salem 1 and Millstone 2.

Westinghouse processed Change Control #9410 on October 1, 1979, and identified corrective actions related to procedural control or to design changes in the safety system actuation circuitry.

b. The inspector reviewed the following documentation from Georgia Power related to requests for assistance from Westinghouse and follow-up on corrective actions to resolve concerns identified in IE Circular 78-19:

(1) GP-3191, NS-PL-6470, response to IE Circular 78-19, July 13, 1979.

(2) BW-2555, request for Westinghouse action related to IE Circular 78-19, April 18, 1979.

(3) BW-2494, response to IE Circular 78-19, July 10, 1979.

(4) GB-1888, request for assistance Re:IE Circular 78-19, January 9, 1979.

c. The inspector determined that Georgia Power has selected the corrective action related to a redesign of the safety system actuation circuitry to ensure that safety injection signals are not blocked coincidentally with the block of the high radiation signal. This corrective action was transmitted to the applicable Project Manager on January 18, 1980, for his review and approval.

7. Undetectable Failure in the Engineered Safety Features Actuation System

This item involves the non-testability of the P-4 (Permissive) contacts which could result in an undetectable failure and subsequent prevention of the normal mode of resetting and blocking safety injection and consequently alter the sequence of switchover operations from injection to the recirculation phase of accident recovery.

a. This item was reported to the NRC by Westinghouse on November 7, 1979, as a 10 CFR Part 21 substantial safety hazard for operating plants, and as a 10 CFR 50.55(e) significant deficiency for plants under construction (NS-TMA-2150). This report generically addresses those operating and non-operating plants in the United States which are affected, and recommends corrective action to be implemented at each.

b. This item was identified by Westinghouse and evaluated by the Safety Review Committee on November 6, 1979 (NS-RAW-156, ID-79-149-Block of Safety Injection-Meeting Minutes, 11/12/79).

Notification to affected utilities was made by Westinghouse on November 8, 1979 (AEP-79-39). Additional information was supplied related to undetectable failure in the Engineered Safety Features Actuation System on November 26, 1979 (AEP-79-44).

8. Lower Reactor Vessel Head Insulation Support

This item relates to the movement of the reactor vessel insulation support brackets with respect to the instrument guide tubes during a postulated seismic event. In particular, a field modification, during the installation of the insulation support brackets, allowed the support brackets to be clamped directly to the instrumentation guide tubes. During a seismic event, this modification could have resulted in forces being imposed on the guide tubes greater than those already analyzed.

a. This item was originally reported by North Anna 1 as a 10 CFR 50.55(e) construction deficiency on November 8, 1978.

The NRC issued IE Information Notice No. 79-11, Lower Reactor Vessel Head Insulation Support Problem, on May 7, 1979.

TVA reported this item as a 10 CFR 50.55(e) construction deficiency on June 4, 1979, for Sequoyah 1 and 2 and Watts Bar 1 and 2.

b. The inspector determined that this item has been evaluated by Westinghouse and appears to be unique to North Anna 1, in that, the field modification for the lower reactor vessel head insulation support was made by clamping the insulation support brackets to the instrument guide tubes. This item is therefore not generic to other domestic Westinghouse plants. The clamps will be removed at North Anna 1 and the framework will be modified. TVA most probably reacted to the IE Information Notice and should not have reported this item.

9. Findings

In this area of the inspection, no items of non-compliance, deviations, or unresolved items were identified.

C. Exit Meeting

An exit meeting was held with management representatives on January 24, 1980, at the conclusion of the inspection. Those persons noted by an asterisk in the Details Sections of this report were in attendance. The inspectors discussed the scope of this inspection and the details of the findings identified. Management representatives acknowledged the comments of the inspectors with respect to these items discussed.

DETAILS SECTION II

(Prepared by R. H. Brickley)

A. Persons Contacted

*T. M. Anderson, Manager, Nuclear Safety
R. A. Loose, Manager, Safegards Systems
*P. T. McManus, Senior Quality Engineer
R. R. Oft, Engineer, Safegards Systems
*R. A. Wiesemann, Manager, Regulatory and Legislative Affairs

*Denotes those present at the exit meeting.

B. Steam Generator Water Level Deficiency

1. Background

On October 5, 1979, the Public Service Electric and Gas Company (Salem, Unit No. 2) submitted a written 10 CFR 50.55(e) report to Region I regarding the subject item. They had been notified by Westinghouse that a potential safety problem existed with the heatup of the steam generator level measurement reference legs during accident conditions. Pipe breaks inside the containment resulting in elevated containment ambient temperatures could cause heatup of the steam generator level reference legs. This would result in a decrease in water column density and an increase in the indicated steam generator water level. The actual water level will be lower than the level indicated by the instruments. The erroneous indication of level could result in delayed protection system actuation (reactor trip and auxiliary feedwater initiation) and could affect operator response for post-accident recovery.

2. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the Westinghouse evaluation of this item to determine that a proper evaluation was performed.

b. Determine if this item is generic or plant unique (Salem Unit No. 2)

c. Determine that this item was properly reported to the NRC.

3. Method of Accomplishment

The preceding objectives were accomplished by an examination of the Safety Review Committee (SRC) records (ID-79-143) consisting of:

a. Westinghouse memo (Potential Steam Generator Water Level Measurement System Errors) dated June 13, 1979.

b. Westinghouse letter No. NS-TMA-2104 (Steam Generator Water Level) dated June 22, 1979, notifying the NRC of a potential substantial safety hazard per 10 CFR 21.

c. IE Bulletin No. 79-21 (Temperature Effects on Level Measurements) dated August 13, 1979.

4. Findings

a. This item was identified and processed per the Water Reactor Divisions (WRD) Policy/Procedure No. WRD-OPR-19.0 (Identification and Reporting of Substantial Safety Hazards, Significant Deficiencies, and Unreviewed Safety Questions) Revision 0, dated 7/19/79. Note: This policy/procedure was originally issued as OPR-WRD-600-1.

b. The item was determined to be a potentially substantial safety hazard generic to Westinghouse plants. The Commission was notified via telephone on June 21, 1979, and 10 CFR 21 report on June 22, 1979. This notification included their recomendations for corrective actions i.e. corrections to the indicated steam generator water level, low water level protection system setpoints, and emergency operating procedures.

c. There were no noncompliance, deviation, unresolved, or follow-up items identified.

C. Barton Differential Pressure Transmitters

1. Background

A reference to this item, found during the examination of the SRC Records (ID-79-143), was the basis for this area of the inspection.

2. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the Westinghouse evaluation of this item to determine that a proper evaluation was performed.

b. Determine if this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

3. Method of Accomplishment

The preceding objectives were accomplished by an examination of the SRC records (ID-79-142) consisting of:

a. Westinghouse memo No. E-PCB-2273 (Potential Deficiency, Barton Lot No. 1) dated May 21, 1979.

b. Minutes of the June 7, 1979 meeting of the SRC.

c. Westinghouse memo No. E-PCB-2321 (Potential Deficiency, Barton Lot No. 1) dated June 8, 1977.

d. Westinghouse letter No. NS-TMA-2098 dated June 11, 1979, notifying NRC of a potential substantial safety hazard per 10 CFR 21.

4. Findings

a. This item was identified and processed per WRD Policy/Procedure No. WRD-OPR-19.0.

b. This item involved a deficiency that had been identified in the procedure used to check the performance characteristics of the Barton Lot 1 transmitters prior to shipment. Westinghouse determined that a maximum of forty-eight (48) differential pressure transmitters delivered to the plants (identified in the letter) for use in the Steam Generator Level (narrow range) function may exhibit a positive inaccuracy in excess of the Westinghouse specification of +10 percent. Depending on the magnitude of the inaccuracy of each transmitter and their disposition within the plant, the protective actions that rely on the Steam Generator Level (narrow range) initiation signal could be impaired. Westinghouse requested that the applicable Licensees return the transmitters to Barton for checking and circuit modification to bring the performance within specification.

c. This item was found to be applicable to McGuire 1, Farley 1, Diablo 1 and 2, Salem 2, Cook 2, Sequoyah 1, and Watts Bar 1.

d. The Commission was first notified via telephone on June 8, 1979, followed by a 10 CFR 21 letter of notification No. NS-TMA-2098 dated June 11, 1979.

e. There were no noncompliance, deviation, unresolved, or follow-up items identified.

D. Deficiency in Net Positive Section Head (NPSH) for RHR Pumps

1. Background

On October 4, 1979, the Public Service Electric and Gas Company (Salem, Unit No. 2) submitted a written 10 CFR 50.55(e) to NRC Region I regarding the subject item. During tests of the RHR pumps to establish the maximum flow for the worst hydraulic configuration and to evaluate the available NPSH at that flow, the data indicated that if the throttle valves were open to 100% the design runout flow of 4500 gpm would be exceeded. Their corrective action was to increase the resistance on the discharge side of the pumps by changing the orifices on the flow elements upstream and downstream of the RHR heat exchangers. The licensee further reported that Westinghouse had evaluated the modified system for safety injection and concluded it to be above the reference performance contained in the FSAR.

2. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the Westinghouse evaluation of this item to determine that a proper evaluation was performed.

b. Determine if this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

3. Method of Accomplishment

The objectives were accomplished by an examination of calculation No. SD/SS-BU-045C (Flow Orifice Increased Pressure Drops) dated September 13, 1979.

4. Findings

a. Westinghouse has evaluated the modified system and demonstrated that it meets the performance requirements contained in the FSAR.

b. Westinghouse engineers attributed the difference between actual and calculated flow to the conservation built into the values obtained from the Crane handbook.

c. This item was determined to be unique to the Salem plants and therefore properly reportable under 10 CFR 50.55(e).

E. Deficiencies in Printed Circuit Process Cards

1. Background

On August 23 and 28, 1979, NRC Region II was notified by their licensees (Summer and Farley I) that Westinghouse had identified deficiencies in process cards that could result in altering the limiting condition setpoints during a seismic induced circuit malfunction.

2. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the Westinghouse evaluation of this item to determine that a proper evaluation was performed.

b. Determine if this item is generic or plant unique.

c. Determine that this item was properly reported to the Commission.

3. Method of Accomplishment

The preceding objectives were accomplished by an examination of the SRC records (ID-79-145 & ID-79-146) consisting of:

a. Minutes of the August 21, 1979 SRC meeting concerning the 7300 series process I&C card failures.

b. Westinghouse letter No. NS-TMA-2124 (Westinghouse 7300 Series Process Control System) dated August 23, 1979, to the NRC notifying them of a potential substantial safety hazard per 10 CFR 21.

4. Findings

a. The Westinghouse assessment of field reports on the 7300 Series Process Control System identified two (2) technical problems which could exist. The first involved a circuit component in protection system comparators (bistables) which were observed to have an abnormal failure rate in that application. The record involved the potential for a seismic-induced circuit malfunction which could alter limiting setpoints for initiating safety action by the 7300 Series system.

b. This item was identified and processed per WRD Policy/Procedure No. WRD-OPR-19.0.

c. The Commission was notified of this item by letter No. NS-TMA-2124 on August 23, 1979. This letter identified all affected plants and the corrective actions to be taken.

d. There were no noncompliance, deviation, unresolved, or follow-up items identified.

F. Raytheon RC747D Integrated Circuit Chips

1. Background

A reference to this item, found during the examination of SRC records (ID-79-145 & ID-79-146) was the basis for this area of the inspection.

2. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the Westinghouse evaluation of this item to determine that a proper evaluation was performed.

b. Determine if this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

3. Method of Accomplishment

The preceding objectives were accomplished by an examination of SRC records (ID-77-127) consisting of:

a. Westinghouse letter No. APW-A-4713 (7300 Process Control system Potential Significant Deficiency) dated December 12, 1977, to a licensee (Farley).

b. Westinghouse memo No. NS-CE-1630 (WRD Safety Review Committee Meeting) dated December 12, 1977.

c. In addition to the above, the following documents were presented to the inspector by Westinghouse management:

(1) Westinghouse letter NAW-3146 (Process I&C Deficiency) dated December 9, 1977, to a licensee (VEPCO).

(2) An NRC internal memorandum (Inquiry Into Recent Reactor Event Practice by VEPCO) from the Director, I&E to the Director, NRR dated December 12, 1977. The enclosure to this memo was found to be Investigation Report Nos. 50-338/77-59 and 50-339/77-38 complete with enclosures.

(3) A Southern Services Company letter No. GLF-NS-182 (7300 Process Control System) dated February 9, 1978, to Alabama Power Company (Farley 2) recommending a 10 CFR 50.55(e) report be submitted. A copy of the draft report was attached.

4. Findings

a. This item was found to deal with failures detected as a result of Westinghouse tests of amplifiers employing integrated circuit chips identified by Raytheon as RC747D. The failure mechanism results from conductive particles that are dislodged inside some of the amplifiers during seismic excitation. The SRC evaluation concluded that the presence of conductor particles created the potential for a common mode failure mechanism during a seismic event. The affected amplifiers are used by numerous circuit boards in the reactor protection, control, and engineered safegards systems. Chips which fail due to internal short circuits or other causes would be detected during routine, periodic, on-line testing of the 7300 equipment. Testing, however, does not preclude the possibility that failures, due to dislodged conductive particles, could significantly reduce the probability of reactor trip and safeguards actuation during seismic events.

The affected Raytheon amplifiers are used primarily in 7300 systems supplied as original equipment by WISD during the period 1973 through 1975. Any plant employing a 7300 system fabricated during this period is affected. Any plant which as been supplied with spare 7300 circuit boards fabricated during this period could also be affected. 7300 systems fabricated prior to 1973 employ Fairchild integrated circuit amplifiers. 7300 systems fabricated after 1975 employ Motorola integrated circuit amplifiers. The Fairchild and Motorola amplifiers are not subject to the conductive particle problem.

The Committee concluded that for plants with Fairchild and Motorola amplifiers there was no safety problem with the original equipment as shipped. However, if original equipment has been replaced by spare parts containing the Raytheon amplifiers, depending on the number of replacements and the specific circuits involved, there could exist a significant deficiency for affected plants under construction or an unreviewed safety question or substantial safety hazard for affected operating plants. For those plants with original equipment containing the Raytheon amplifiers (no domestic plants in this category are operating), the committee concluded that a significant deficiency exists. Accordingly, the committee recommended notification of affected customers and that the NRC notification to the extent appropriate be via affected customers.

b. The plants where 7300 systems using the Raytheon amplifiers were supplied as original equipment are Byron 1 and 2, Farley 2, and McGuire 1 and 2. In addition to these plants those which may have spares with Raytheon amplifiers are North Anna 1 and 2, Farley 1, Braidwood 1 and 2, Millstone 3, Tyrone, and Summer.

c. Noncompliances

None identified.

d. Deviations

None identified.

e. Unresolved Item

Sufficient documentation was not examined to verify that all customers receiving defective Raytheon RC 747D integrated circuit chips had been notified.

f. Follow-up Items

None identified.