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Docket Nos. 50-348, 50-364
License Nos. NPF-2, NPF-8

Alabama Power Company
ATTN: Mr. R. P. McDonald
Senior Vice President
P. O. Box 2641
Birmingham, AL 35291-0400

Gentlemen:

SUBJECT: ELECTRIC EQUIPMENT ENVIRONMENTAL QUALIFICATION ISSUES
(NRC INSPECTION REPORT NOS. 50-348/87-30 AND 50-364/87-30)

This letter refers to the Management Meeting held in the Region II office, Atlanta, Georgia on November 25, 1987. The issues discussed at this conference related to environmental qualification (EQ) of electrical equipment. A meeting summary, a list of attendees, and a copy of the handout are enclosed.

It is our opinion that this meeting was beneficial in helping to bring a speedy resolution to a very complicated EQ operational issue. It also provided for a better understanding of the inspection findings and the status of your corrective actions.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this matter, please contact us.

Sincerely,

/s/

J. Nelson Grace
Regional Administrator

Enclosures:
1. Meeting Summary
2. List of Attendees
3. Handout:
   (a) Justification for Continued Operation (JCO) Unit 1 - Technical Blocks Used in Instrument Circuits
   (b) Raychem/Chico Environmental Seal Qualification

cc w/encls: (See page 2)

8802030037 880122
PDR ADOCK 05000348
Q PDR

IE01

cc w/encls:
W. O. Whitt, Executive Vice President
J. D. Woodard, General Manager - Nuclear Plant
W. G. Hairston, III, Vice President - Nuclear Support
J. W. McGowan, Manager-Safety Audit and Engineering Review
J. K. Osterholtz, Supervisor-Safety Audit and Engineering Review

bcc w/encls:
NRC Resident Inspector
E. Reeves, Project Manager, NRR
DRS Technical Assistant
Document Control Desk
State of Alabama
E. Merschoff, RII
B. Grimes, NRR
G. Lainas, NRR
U. Potapovs, NRR
M. Jacobus, Sandia National Laboratories

| RII | RII | RII | RII | RII | RII | RII |
|---|---|---|---|---|---|---|
| NMerriweather:lb | TEConlon | ARHerdt | AGibson | HDance | LReyes | MLErnst |
| 12/30/87 | 12/30/87 | 12/31/87 | 1/6/88 | 1/12/88 | 1/19/88 | 1/21/88 |

## ENCLOSURE 1

### Meeting Summary

On November 25, 1987, representatives of the Alabama Power Company (APCO) met with NRC Region II personnel in Atlanta, Georgia to discuss (1) the issues of equipment qualification (EQ) that derived from the recent inspection, (2) any ramifications of these issues on continued operation of Unit 1, and (3) corrective actions taken before restart of Unit 2. The list of those who attended the meeting is Enclosure 2.

Following opening remarks given by M. L. Ernst, Region II, Deputy Regional Administrator, APCO gave a presentation which addressed the specific concerns that the NRC had requested. The majority of the meeting concerned the licensee's justification to allow continued operation of Farley Unit 1 with terminal blocks installed in various instrument loops inside containment. The licensee's previous position had been that these terminal blocks were qualified, but the staff disagreed. In the meeting, the licensee then took the position that the terminal blocks were qualifiable, and also presented a JCO to justify continued operation based on a combination of EQ data and a semi-quantitative assessment of containment temperatures under realistically bounding accident conditions. The basis for the licensee's positions is outlined in their handout. (Enclosure 3.a.).

The staff's conclusions and the licensee's commitments are discussed in a confirmation of action letter dated December 2, 1987, which is included herein as Attachment A.

Other issues which were discussed at the meeting related to Raychem/Chico seals (Enclosure 3.b.), Raychem Stilan cable, ASCO solenoid valves, Gems level transmitters and Limitorque motor operated valves (MOVs). The licensee made the following comments on these items:

a. Raychem/Chico seals are considered qualified to NUREG 0588 Cat. II.
b. The Reactor Vessel Head Vent Valves cable entrance seals were replaced with the Raychem/Chico Seal design on Unit 2.
c. Raychem Stilan Cables were replaced on Unit 2 and will be changed out on Unit 1.
d. The Gems level transmitters have been replaced on Unit 2.
e. The qualified life for ASCO solenoid valves was recalculated to greater than 11 years.
f. T-drains will be installed on those Limitorque MOVs, where allowable.
g. Terminal Blocks in Unit 2 Limitorque MOVs have been inspected and a JCO was documented for all Unit 1 Limitorque MOVs that use terminal blocks.

The staff commented that the licensee's presentation on the Raychem/Chico Seal did not demonstrate qualification but it could be used as a basis to develop a justification for continued operations.

The meeting was closed with final comments from Mr. M. L. Ernst.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

~~Upright~~

ATTACHMENT A

DEC 2 1987

Docket No. 50-348
License No. NPF-2

Alabama Power Company
ATTN: Mr. R. P. McDonald
Senior Vice President
P. O. Box 2641
Birmingham, AL 35291-0400

Gentlemen:

SUBJECT: CONFIRMATION OF ACTION - DOCKET NO. 50-348

This refers to the Management Meeting held in the Region II Office, Atlanta, Georgia on November 25, 1987. This meeting was held to discuss the issues of equipment qualification (EQ) that derived from the recent inspection and any ramifications of these issues on continued operation of Unit 1 and corrective actions taken before restart of Unit 2.

The licensee stated at the outset of the meeting that, except for justifications for continued operation (JCOs) on grease and lubricants, all EQ discrepancies identified on Unit 2 affecting operability will be fixed prior to Unit 2 startup. Since the meeting, the Region issued a letter (November 30, 1987) permitting Unit 2 startup with one additional outstanding EQ issue regarding Chico/Raychem seals, which is to be resolved by December 2, 1987.

The licensee's position in the meeting regarding Unit 1 focused principally on terminal blocks installed inside containment in various instrument loops. The licensee's previous position had been that these terminal blocks were qualified, but the staff disagreed. In the meeting, the licensee then took the position that the terminal blocks were qualifiable, and also presented a JCO to justify continued operation based on a combination of EQ data and a semi-quantitative assessment of containment temperatures under realistically bounding accident conditions. The licensee also stated that if safety systems initiated early in the accident sequences of concern were allowed to operate, design conditions would not be exceeded. However, the licensee also stated that, using current emergency operating procedures, operator action response to erroneous signals could result in inappropriate actions. Such erroneous signals could occur, if the containment temperature were to exceed that for which the terminal blocks are qualified. After review of the data presented by the licensee, the staff acknowledged that there is disparity in EQ test data for like and different terminal blocks and differences in interpretation of the EQ test data to be applied to Farley. The staff also acknowledged that while the licensee's operability argument had some merit, it was largely based on qualitative assumptions and contained some elements of nonconservatism.

~~8712080110~~ 3pp.

In view of the staff's position, Alabama Power Company (APC) made the following commitments during the meeting, as clarified by telephone calls between C. W. Hehl of the Region II staff and W. G. Hairston of APC on November 30:

1. Once Unit 2 is stabilized at power, but no later than December 9, 1987, initiate an orderly shutdown of Unit 1.

2. In the interim, APC will increase the awareness of shift supervisors and Shift Technical Advisors to the possibility of inaccurate data from instruments located inside containment in the event of a large loss of coolant or steam line break accident. APC will also thoroughly train Shift Technical Advisors of the need to monitor diverse parameters in the event of a large loss of coolant or steam line break accident to assure that inappropriate actions are not taken by operators based on potential inaccurate instrument readings. Further, one of the two STAs on shift will maintain presence in the Control Room.

3. Effect repairs on environmental qualification deficiencies associated with Instrument Terminal Boards and Head Vent prior to restart of Unit 1.

4. Walkdown Unit 1 containment during the spring 1988 refueling outage in a timely manner to identify deficiencies between as-found and as-designed splices and other types of EQ deficiencies found during the recent Unit 2 walkdowns. This walkdown will include all V-splices and a representative sample of other systems and components with field wiring connections to determine if other deficiencies exist.

5. Deficiencies identified shall be repaired on Unit 1 at least to the same extent that repairs had been made to Unit 2 EQ deficiencies. Such repairs shall be made prior to plant startup following the spring 1988 outage.

6. Evaluate operability issues on Unit 1 on any unrepaired deficiencies prior to startup.

7. Plant startup of Unit 1 from the 1988 refueling outage shall not occur without prior concurrence by NRC.

It is the staff's judgment that the Farley terminal blocks might possibly pass a qualification test and that the temperatures at the terminal blocks during a large loss of coolant or steam line break accident might not exceed the temperatures for which the blocks could be qualified. Also, there is a small likelihood of such accidents, and the licensee's compensating actions should enhance proper operator action should instrument inaccuracy occur during such a design basis event. Therefore, we believe that APCs commitments present a reasonable and timely resolution of the issues of environmental qualification of equipment for Unit 1 and will provide reasonable assurance of continued safe operation of the Farley Nuclear Plant Unit 1 for the interim period.

DEC 02 1987



This confirmation of Action (CAL) letter supercedes our CAL of October 6, 1987.

If your understanding of these matters differs from the above, please advise us promptly.

Sincerely,

J. N. Grace

J. Nelson Grace
Regional Administrator

CAL 50-348-87-02

cc: W. O. Whitt, Executive Vice President
J. D. Woodard, General Manager - Nuclear Plant
W. G. Hairston, III, Vice President - Nuclear Support
J. W. McGowan, Manager - Safety Audit and Engineering Review
J. K. Osterholtz, Supervisor - Safety Audit and Engineering Review

ENCLOSURE 2

List of Attendees

Licensee: Alabama Power Company
Facility: Farley Nuclear Plant Units 1 and 2
Location: NRC Region II Office
Atlanta, GA

Date: November 25, 1987
IR No: 50-348, 364/87-30

Alabama Power Company (APCO)

R. P. McDonald, Senior Vice President, APCO
W. G. Hairston, III, Vice President Nuclear Support, APCO
J. D. Woodard, General Manager, Nuclear Plant APCO
W. B. Shipman, Assistant Plant Manager, APCO
R. Berryhill, Systems Performance Manager, APCO
J. McGowan, Manager, Safety Audit and Engineering Review, APCO
J. E. Garlington, Manager, Engineering and Licensing (NEL), APCO
D. H. Jones, Supervisor, Design Support, APCO
D. McKinney, Supervisor, Licensing, APCO
B. S. Monty, Manager, Operational Safeguards, Westinghouse
R. W. Trozzo, Senior Engineer - Nuclear Safety, Westinghouse
P. Dibenedetto, EQ Consultant, DBA
J. Love, Project Engineer, Bechtel

US NRC Region II

M. L. Ernst, Deputy Regional Administrator
A. F. Gibson, Director, Division of Reactor Safety (DRS)
E. W. Merschoff, Deputy Director, DRS
C. W. Hehl, Deputy Director, Division of Reactor Projects (DRP)
A. R. Herdt, Chief, Engineering Branch, DRS
D. M. Verrelli, Chief, Projects Branch 1, DRP
H. C. Dance, Chief, Project Section 1B, DRP
T. E. Conlon, Chief, Plant Systems Section, DRS
P. Fredrickson, Chief, Project Section 1A, DRP
R. J. Goddard, Regional Counsel
L. P. Modenos, Project Engineer, DRP
N. Merriweather, Reactor Inspector, DRS
C. Smith, Reactor Inspector, DRS
A. B. Ruff, Reactor Inspector, DRS
P. A. Taylor, Reactor Inspector, DRS
S. J. Vias, Project Engineer, DRP
W. S. Little, Acting Deputy Director Regional Inspection, TVA
M. D. Hunt, Reactor Inspector, DRS

NRC Headquarters

B. Grimes, Deputy Director, Division of Reactor Inspection and Safeguards, NRR
G. Lainas, Assistant Director for Region II Reactors
U. Potapovs, Chief, Special Projects Inspection Section
M. Jacobus, Engineer, Sandia National Laboratories

Intracompany
Correspondence

ENCLOSURE 3.(a)

Alabama Power

WS 87-0351
Gen. 1193

Subject: Justification For Continued Operation (JCO) Unit 1-Terminal Blocks Used In Instrument Circuit

Date: NOV 24 1987

To: Mr. J. D. Woodard

From: W. G. Hairston, III
At: Vice President, Nuclear Generation

Enclosed is a justification to allow continued operation of Farley Unit 1 with terminal blocks installed in various instrument loops. A copy of this JCO should be placed in the EQ Central File under States, GE and Foxboro terminal blocks.

If you have any questions, please advise.

W G. Hairston III
W. G. Hairston, III

WGH,III/BHW:dst-D72

cc: File: A-3028-JCO
A-5001 IEB 79-01B

Justification for Continued Operation
J. M. Farley - Unit 1
Terminal Blocks Used In Instrument Circuits

## I. BACKGROUND

The qualification of the Farley Nuclear Plant Terminal Blocks used in instrument circuits was based on type test information for the States ZWM Terminal Blocks, the GE CR 151B Terminal Blocks, and the Foxboro Terminal Blocks. Each terminal block tested was identical to that installed in the Farley Nuclear Plants. The terminal blocks were tested under simulated LOCA conditions in a configuration similar to that installed at FNP. Each test resulted in the terminal block successfully performing the intended function. However, although these tests substantiate the acceptability of using terminal blocks under LOCA conditions, the performance parameters that would additionally support their acceptability for use in FNP instrument circuits were not measured. On the basis of the 10CFR50.49 provision that permits type test plus analysis for establishing qualification, an analysis was performed to demonstrate that the FNP terminal blocks could have performed as intended for the instrument application. The analysis demonstrated similarity by size, shape, and function to a terminal block that was type tested under similar FNP LOCA conditions where insulation resistence (IR) was measured to determine leakage current. The analysis further assumed, based on review of the Sandia NUREG/CR-3814 report that the input or change in insulation resistance was attributable to a surface film mechanism and not material dependant. The corresponding values recorded during the test of the similar terminal block (Conax Test Report IPS-107, Connectron Terminal Block) provided a worst case IR value of $3 \times 10^7$ ohms. Allowance of further margin was provided by accepting a lower value of insulation resistance (i.e., $1 \times 10^7$ ohms) for use and input into the FNP setpoint analysis for loop accuracy. (Reference WCAP-11658, Evaluation of the Impact of Cable and Terminal Block Leakage on RPS/ESFAS and ERG Setpoints November 13, 1987). The $1 \times 10^7$ ohms insulation resistance was provided to Westinghouse for all terminal blocks used in FNP instrument circuits.

A review conducted by the NRC during the week of November 16 through 20th indicated that the technical analysis approach used to justify the $1 \times 10^7$ ohms insulation resistance value was not acceptable to the NRC Staff. APCo believes that the methodology employed for the analysis along with the resulting values are technically sound and justified. However, to further exemplify the amount of conservation built into the setpoint analysis, additional reviews and studies were performed.

## II. EVALUATION

A thorough review of the Sandia NUREG report was performed which resulted in confirmation of basic assumptions such as the insensitivity of the terminal blocks to chemical spray, the lack of surface film dependancy on roughness, and the recovery of IR's as temperature is diminished. Additional discussion is provided in Attachment 1 to this report. As explained in Attachment 1, correlation of the Sandia test results to the post accident performance of terminal blocks at FNP can not be made in a quantitative manner.

The previous evaluation of the impact of cable and terminal block leakage on RPS/ESFAS and ERP setpoints (Ref. WCAP-11658, November 13, 1987) considered a conservative value of $1 \times 10^7$ ohms for terminal block IR and, combined with other contributors to channel inaccuracy, confirmed that the RPS/ESFAS functions will occur as required in the plant safety analysis. Furthermore, the use of existing ERP setpoints (without revision) was confirmed to not impact plant safety. At the time of reactor trip and during post accident monitoring, there were no uncertainty increases which could cause the operator to be mislead into performing inappropriate actions. In view of the continuing concerns raised by the NRC regarding the terminal block insulation resistance values currently demonstrated in the FNP EQ documentation and used in instrument inaccuracy studies, an evaluation has been performed to assess the impact of reduced IR values on the ability to achieve and maintain safe shutdown following design basis events. The results of this evaluation are described in Attachment 2.

The evaluation described in Attachment 2 considered the postulated design basis events of large and small break LOCA and secondary pipe breaks. A minimum set of safe shutdown instruments and their functions, potentially exposed to a harsh environment were identified. The evaluation determined that if a terminal block IR value of $5 \times 10^5$ ohms were conservatively assumed as the worst case value for that minimum set of instruments, the resulting instrument inaccuracy will allow the current ERP values to be used without change.

Terminal block testing performed by Sandia National Laboratories (SNL) is documented in NUREG/CR-3416. As discussed in Attachment 1, correlation of the Sandia test results to the post accident performance of terminal blocks at FNP can not be directly made. However, in recognition of the concerns that the Sandia tests have introduced, an evaluation was made of design basis LOCA and secondary pipe break using IR values derived from the Sandia results. Figure 1 represents a correlation between temperature and IR conservatively assuming a logarithmic relationship between temperature and IR. This data is based on IR values for GE EB25 terminal blocks measured at 175°C and 95°C. Additional discussion on the relationship of IR to temperature is contained in Attachment 3. The methodology employed by Attachment 2 was to determine the containment temperature at which the IR value would decrease below the value of $5 \times 10^5$ ohms. At values of $5 \times 10^5$ ohms and above the operator can use his instruments with confidence under the existing ERP's and setpoints. Having determined this containment temperature, the FNP temperature profile is used to define the periods of time when IR is below this threshold value, thereby defining the periods during DBE's when inaccuracy would be postulated to be greater than that accounted for in the ERP's. The results are shown in Figure 3.0-1 of Attachment 2. This period of interest occurs at a time when no operator action is required based on instruments exposed to the postulated harsh environment. For large and small LOCA, no mitigative or recovery operator actions are required using instrumentation in a harsh environment. For secondary breaks, safety injection termination (the required manual operator recovery action) will occur after the instrument accuracy returns to an acceptable value. The onset of excessive instrument inaccuracies as shown in Figure 3.0-1 is not expected during a DBA since the following conservative assumptions were considered:

1. The test profile shown in Figure 1 of Attachment 3, used to obtain the IR values assumed in Figure 1 greatly exceed the maximum calculated design basis LOCA/MSLB temperature profile for FNP.

2. The physical configuration of Phase I specimens in the Sandia test produced more severe conditions than would occur at FNP. The conduit was routed up the exterior of the enclosure and terminated in the test chamber approximately 12 inches below the steam inlet port and the spray header. Neither end of the conduit was sealed. (See Attachment 1.)

3. Sufficient test data exists to indicate that #12 AWG conductors will exhibit lower IR values than smaller #16 AWG conductors with the same insulation system. The Sandia testing used #12 AWG cables whereas #16 AWG is used in FNP field cables for RTD and transmitter applications. (See Attachment 1.)

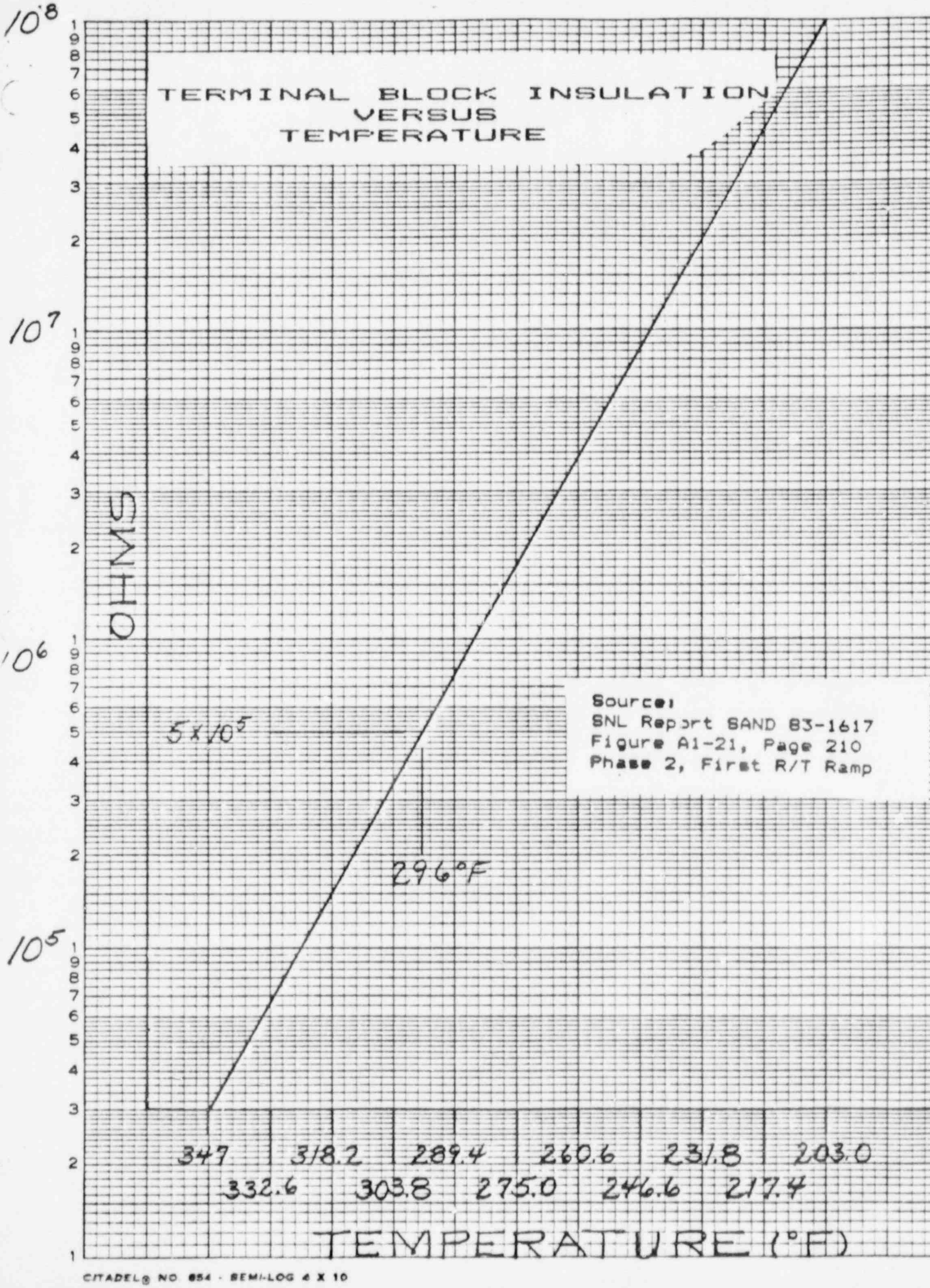
4. The containment temperature profile assumed is derived from worst case assumptions described in FSAR Chapter 6.2 including 102% power, minimum ESF, and only one containment cooler. The profile which would result from more realistic assumptions would be significantly lower.

5. The minimum values of IR and corresponding high leakage currents recorded in the referenced SNL test results are conservative, and are not representative of values that would be expected at FNP during LOCA/MSLB design basis events. Minimum values of terminal block IR values higher than those recorded in the SNL report are supported by CONAX Text Report IPS-107, and Wyle Report Nos. 17775-1 and 17733-1 for MSLB/LOCA temperatures relevant to FNP. (See Attachment 3.)

## III. CONCLUSION

Based on the above, Alabama Power Company concludes that there is reasonable assurance that the instrument loops will perform their safety function when called upon to mitigate the accident for which they are needed. However, to further remove the point of contention regarding terminal block performance and thereby increase the margin of the Westinghouse setpoint analysis, APCo will replace the terminal blocks of concern in Unit 2, during the fifth refueling outage, with qualified splices not relying on terminal blocks and APCo will take the same measure for Unit 1 prior to startup from the eighth refueling outage, currently scheduled for March 1988.

FIGURE 1



TERMINAL BLOCK INSULATION
VERSUS
TEMPERATURE
OHMS
10^8
10^7
10^6
10^5
5x10^5
296°F
Source:
SNL Report SAND 83-1617
Figure A1-21, Page 210
Phase 2, First R/T Ramp
347
332.6
318.2
303.8
289.4
275.0
260.6
246.6
231.8
217.4
203.0
TEMPERATURE (°F)
CITADEL® NO 654 · SEMI-LOG 4 X 10

ATTACHMENT 1

Additional Clarifications Regarding the Qualification of States NT/ZWM and G.E. CR151B Terminal Blocks at Farley Nuclear Plant (FNP) Units 1 and 2 in Low Voltage RPS/ESFAS and ERP Transmitter and RTD Circuits

QUALIFICATION REQUIREMENTS AND STATUS

States terminal blocks mounted in NEMA 4 enclosures, and G.E. CR151B terminal blocks provided with the G.E. Series 100 electrical penetration assembly terminal boxes were installed in containment safety related instrumentation circuits at FNP during construction. As such these blocks including their performance and installed configuration were required to be and are qualified to the DOR Guidelines for FNP Unit #1 and to NUREG-0588, Cat. 2, for FNP Unit #2. In accordance with 10CFR50.49, Par. K, requalification of this electric equipment is not required.

EFFECTS OF LOCA/MSLB ENVIRONMENT ON TERMINAL BLOCK LEAKAGE CURRENTS AND PERFORMANCE

IE Information Notice No. 84-47 indicated that as a result of testing performed by Sandia National Laboratories (SNL) for the NRC it was shown that a moisture film will form on the surface of terminal blocks during the simulation of LOCA/MSLB events. (Ref: NUREG/CR-3418; SAND83-1617, Printed August 1984. Note that this reference was not provided in IEN 84-47). This film will result in the reduction of insulation resistance between terminal points and ground, and thus will allow some leakage currents to flow to ground. IEN 84-47 further states that the NRC staff recognizes that leakage currents do exist during LOCA/MSLB simulations and that the leakage currents may be of significance in some applications.

No written response to the notice was required, and it was suggested that licensees:

1. Review their facilities to determine if terminal blocks are used in low-voltage applications, such as transmitters and RTD circuits, and

2. Review terminal block qualification documents to ensure that the functional requirements and associated loop accuracy of circuits utilizing terminal blocks will not degrade to an unacceptable level due to the flow of leakage currents that might occur during design basis events.

The notice further stated that the NRC staff considers this review to be part of the on-going activities that licensees are currently undertaking to resolve other environmental deficiencies per 10CFR50.49 deadlines and requirements.

IEN 84-47 indicated that where existing terminal block qualification testing does not provide supporting data for instrumentation leakage currents, the following possible corrective action could be considered:

> Obtain documentation from valid qualification tests already performed with substantiated data for leakage currents, and perform appropriate analysis to demonstrate that acceptable loop accuracy and associated response times for instrument circuits utilizing terminal blocks are being maintained throughout various operating conditions.

Two other possible corrective actions were also stated which involved either additional qualification testing of installed terminal blocks with provisions for continuous monitoring of leakage currents throughout the test with analysis of loop accuracies, or replacement of installed terminal blocks with qualified splices.

FNP EVALUATION OF TERMINAL BLOCK LEAKAGE CURRENTS

States terminal blocks in NEMA 4 enclosures were qualified for FNP Instrumentation and Control circuits inside containment by Wyle Report No. 44354-1. Post LOCA simulation of Insulation Resistance (IR) values were recorded, but no leakage current or IR values were recorded during the LOCA test phase to permit quantification of the surface moisture film leakage currents discussed in IEN

84-47. CR151 and States NT terminal blocks installed in G. E. Series 100 Low-Voltage Instrumentation and Control Penetration NEMA 4 terminal boxes inside containment were qualified for FNP by G.E. as stated in G.E. Qualification Test Summary Report 994-75-011, dated March 27, 1975. This report provides one minimum value for IR associated with LOCA simulation testing of the CR151 and States blocks, but insufficient leakage current or IR values recorded during the LOCA test phase exist to permit quantification of the surface moisture film leakage currents discussed in IEN 84-47.

Due to the lack of data recorded in the DOR Guideline and NUREG-0588 Cat. 2 qualification reports for the FNP States and CR151B terminal blocks installed in NEMA 4 enclosures, a documentation search was conducted to obtain documentation from already performed valid qualification tests of identical or similar terminal blocks which could provide leakage current or IR data recorded during the simulated LOCA steam conditions. Of the test report documents evaluated, including the SNL test documentation upon which IEN 84-47 was based, the most representative test of FNP in containment terminal block and enclosure configurations which provided IR readings during simulated LOCA/MSLB steam conditions was Conax Report No. IPS-107, dated 10/5/73. Minimum IR values contained in this report which were obtained during LOCA simulated steam conditions were reviewed and a conservatively low IR value was provided to Westinghouse for determination of the resulting leakage currents and their affects on RPS/ESFAS and ERP setpoint accuracies.

WCAP-11658 addresses the results of this evaluation , and response to APCo E. Q. Action Items 018 and 067, addresses the methodology used for the selection of the terminal block IR value used in the Westinghouse evaluation.

## BASIS FOR NOT USING SNL IR OR LEAKAGE CURRENT VALUE FOR WCAP-11658 EVALUATION

All the following comments are based on a review of NUREG/CR-3418, SAND83-1617 entitled "Screening Tests of Terminal Block Performance in a Simulated LOCA Environment" printed August 1984 and are in reference to sections of that document (Attachment #1A to this clarification report). It is important to note that only Phase I testing was performed on G.E. CR151B (Manufacturer I, Model B) and States ZWM (Manufacturer III, Model D) terminal blocks as shown in Table 1, Pg. 12.

o Environmental Test Temperature and Pressure Profiles - As shown in Figure 1, Pg. 8, the test temperature and pressure peaks as well as profile durations greatly exceeded the maximum calculated

DBE LOCA/MSLB surface temperature conditions for FNP in containment terminal block applications. As stated in the last paragraph on Pg. 52 of Sect. 4.3.4, "Terminal blocks 6,11, and 12 (States ZWM) experienced a temperature effect. Their inter-terminal barrier softened almost to the liquid melt point, and flowed from between the terminals. The melted material covered some of the lower posts of the terminals, encasing the wires and drooping below the temrinal block in large globules. Surprizingly, as Figure 20 shows, the terminal-to-terminal insulation resistances for terminal blocks 6, 11, and 12 were among the highest measured. We have no reasonable hypothesis to explain this behavior. We can speculate that the phase change of the inter-terminal barrier material prevented in someway the formation of a continuous film between terminals, or that changing geometry somehow affected the process of conduction between adjacent terminals". Geometrical changes of the inter-terminal barrier occured in Wyle Test 44354-1, but complete melting did not occur.

o No chemical spray was introduced in Phase I LOCA Testing. (However, Section 5.5, Pg. 126 of the conclusion states that little change in the moisture film conductivity may be expected as a result of chemical spray and therefore, chemical spray would appear to not be a significant issue.)

o Physical Configuration of Phase I Specimens - Three 6-pole CR151B and three 6-pole States ZWM terminal blocks were all mounted vertically in the same NEMA 4 enclosure (Enclosure 2) as shown in Figure 4, Pg. 11. Cables were brought into the side of the enclosure through 3/4 inch diameter liquid tight metal hose using elbow conduit terminators to penetrate the NEMA 4 enclosure walls. The conduit was routed up the exterior of the enclosure, and terminated in the test chamber head approximately 12 inches below the steam inlet port and the spray header. Neither end of the conduit ws sealed. (See bottom Pg. 16, and top of Pg. 18.)

All cables used to connect the terminal block test circuitry were #12 AWG, either 1-conductor or 3-conductor. The direct steam jetting exposure into the open conduit from the steam inlet port is not representative of installed instrumentation conduit configurations at FNP, and the use of #12 AWG single conductor and multi-conductor cable is not representative of the FNP installed

ATTACHMENT 3

November 24, 1987

TO: JOHN GARLINGTON

FROM: JESSE LOVE

IR vs TEMPERATURE SUPPORTING INFORMATION FOR JCO

As documented in numerous valid test reports, conducted by Wyle, SNL and other industrial test organizations, electrical cable and terminal blocks exhibit generic characteristics with regard to insulation resistance (IR) versus temperature during simulated LOCA/MSLB test conditions. The generic characteristic is that IR values are inversely proportional to temperature i.e. lower temperature yields higher value of IR. Conversely with regard to leakage current, leakage current is directly proportional to temperature. SNL Report SAND 83-1617 provides numerous data representations which demonstrate this accepted phenomenon.

Figure 1 of Westinghouse letter dated 11/23/87 was made from plots of SAND83-1617 (SNL) Phase II test data for exposure of G.E. EB25 terminal blocks to the SNL Phase II simulated LOCA/MSLB profile (Attached Figure 2, Pg. 9 of SAND83-1617). IR test data for an EB25 block was used from the SNL report as there were no States ZWM, or CR151B blocks tested by SNL in Phase II, and the EB 25 block is similar to these FNP installed blocks. Phase I data which did record leakage currents and IR values for States ZWM and CR151B blocks was not used due to the inaccuracies associated with the SNL electrical test circuitry that measured leakage current values during Phase I testing.

Figure 1

The minimum values of IR and corresponding high leakage currents recorded in the referenced SNL test results are extremely conservative, and are not representative of values that would be expected at FNP during LOCA/MSLB design basis events. Minimum values of terminal block IR values higher than those recorded in the SNL report are supported by CONAX Text Report IPS-107, and Wyle Report No.s 17775-1 and 17733-1 for MSLB/LOCA temperatures relevant to FNP.

instrumentation cable. Installed instrumentation cables at FNP for RTD and transmitter applications are #16 AWG.

Sufficient test data exists which appears to indicate that #12 AWG conductors will exhibit lower IR values than smaller #16 AWG conductors with the same insulation system when exposed to LOCA steam conditions. As the #12 AWG cable is a part of the test circuit and its contribution to IR and leakage currents resulting from steam moisture is included in the terminal block measured data, additional error may have been introduced.

o Electrical Configuration of Phase I Test-(Sect. 3.4, Pg. 10, Figure 10, Sect. 4.1, Pg. 29 and last paragraph Pg. 94).

A serpentine connection of alternate terminal block (TB) poles was used which did not result in the measurement of a unique pole-to-pole resistive path. As stated in Sect. 4.1 "The serpentine connection of the 6-pole terminal blocks actually provided 5 parrallel resistive paths. Each of these paths, indicated $R_1$ through $R_5$ in Figure 16, is in turn a parallel combination of an infinite number of paths, i.e...a surface.*" "In measuring the leakage currents the equivalent resistance of these 5 surfaces is actually measured. Without further data or assumption the individual values of the surface equivalent resistances, $R_1$ through $R_5$ cannot be determined".

Also as stated in Sect. 3.4 "only one ground return path existed for all 12 phase I terminal blocks, 6 blocks per enclosure. For the majority of the Phase I test, all blocks were powered simultaneously, and hence only pole-to-pole leakage current data is relevant".

As stated in Section 4.4.3, Pg. 94, last paragraph, "If the conduction paths were uniformly distributed over the terminal block surface, the differences in wiring between Phase I (serpentine) and Phase II (straight through), would cause the Phase I IR's to be less than the Phase II IRs. This result is a simple consequence of multiple parallel conducting paths. For our experimental configuration there was approximately five times the pretested conducting surface available on the Phase I terminal blocks as compared to the Phase II terminal blocks. Consequently, the insulation resistance for the Phase I terminal blocks could reasonably be expected to be one fifth of the Phase II IR's. Except for the A path of Phase II terminal block 4, the 45Vdc data and the 125 Vdc support the hypothesis of uniformly distributed conduction."

The serpentine test circuitry used to measure the States and CR151B test specimen leakage currents and IR's did not yield direct individual pole-to-pole or pole to ground values of IR during the LOCA steam environment simulation, and are subject to hypothesis in order to arrive at required pole-to-pole values.

o General Applicability of Phase II Test Data - As stated above, no Phase II testing was performed on CR151B or States terminal blocks. The only block tested in Phase II based on present available information which appears to be similar to the CR151B and States blocks with regard to, block material, pole-to-pole spacing, the presence of a barrier between poles and a one-piece non channel mounted block is the G.E. EB25 (Manufacturer I, Model A). It should be noted that Table, 1 Pg. 12, incorrectly states that the States ZWM block is a sectional block. Six EB25 blocks were tested in Phase II. Although, the electrical test circuitry of the Phase II test yields more realistic values of leakage currents and IR's than Phase I test, other electrical test anomolies, and the configuration and environmental test profile are not representative of the installed condition at FNP.

It is interesting to note that the only physical design affects analyzed were related to whether or not the blocks were sectional or one piece as stated in Sect. 4.4.1.3, Pg. 81. No apparent attempt was made to correlate leakage current performance to geometrical considerations such as the presence of barriers and height of blocks with barriers between poles or pole-to-pole spacing. Perhaps the conclusion stated in Sect. 4.4.1.3 that "Figures 34 through 39 show about one to two orders of magnitude difference between the performance of terminal blocks 5, 6, and 12 and the one piece blocks, the one piece blocks being better." is not singularly related to the sectional block design, but to other geometrical considerations. For the Phase II tests, the one piece blocks referenced here are G.E. EB25 blocks which have similar pole spacing to the G.E. CR151B and States ZWM one piece blocks and do possess barriers between poles.

ATTACHMENT 2

Westinghouse Electric Corporation

Power Systems

Nuclear Technology Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

ALA-87-882
Ref: ES 87-1000

November 23, 1987

Mr. W. G. Hairston, III, Vice President
Nuclear Generation
Alabama Power Company
600 North Eighteenth Street
Birmingham, Al 35291-0400

Attn: Mr. J.E. Garlington

Joseph M. Farley Nuclear Plant
Units No. 1 & 2
ERP Information

Dear Mr. Hairston:

Attached is additional information on the report which was generated for Alabama Power Company entitled "Evaluation of the Impact of Cable and Terminal Block Leakage on RPS/ESFAS and ERP Setpoints" dated November 1987. This information was generated as a result of the NRC Equipment Qualification Audit which was held during the week of November 16, 1987.

If you have any additional questions regarding this please contact the project office.

Very truly yours,
WESTINGHOUSE ELECTRIC CORPORATION

G G Knochel
for C. Eichaldinger, Manager
Alabama Project

AJ/DGL/dmr

ATTACHMENT ALA-87-882

The attached table contains a listing of Farley Unit 1 Emergency Response Procedure (ERP) harsh environment instruments, significant safety related functions of each instrument, and time usage factors and diverse instruments for each function. The purpose of the table is to list the instruments potentially subject to a harsh environment for the Farley design basis events. These instruments have an environmental allowance in their calculated uncertainties used in the ERPs. The design basis events are large and small LOCA and secondary system pipe breaks; i.e., steam line and feed line breaks.

A review of this table results in identification of a minimum set of instruments, and their functions, subject to a harsh environment and also necessary for safe shutdown from design basis events. These are RCS Subcooling, Wide Range Pressure, and Narrow Range Steam Generator Water Level. Backup instruments have been identified where available. Other instruments necessary for safe shutdown are located in a mild environment or are not affected by current leakage. Other instruments used in the ERPs are not used to base any required actions within the Farley design basis events or will not cause any actions to be taken detrimental to plant safety if the instument uncertainty exceeds the allowance presently in the Farley ERPs.

For RCS Subcooling, Steam Generator Narrow Range Level and Wide Range Pressure, it is recommended that for Farley Unit 1 that a containment temperature criterion be defined that is indicative of current leakage resistance of less than 5 X $10^5$ ohms. A value of greater than 5 X $10^5$ ohms results in an instrument inaccuracy that will allow the current ERP values to be used by the operator to take action as specified in the ERPs. The temperature or a corresponding containment pressure criterion should be used as guidance to the operator using the ERPs on when to consider that additional error above that already accounted for in the ERPs may exist. Under conditions exceeding these criteria no actions which could reduce the margin of safety, specifically termination of safety injection based on RCS Subcooling or stopping of all auxiliary feedwater based on Steam Generator Narrow Range Level or stopping of RHR pumps based on Wide Range Pressure, should be performed since the errors may exceed those accounted for in the ERPs. After containment conditions have returned to below these criteria the operator can safely resume the use of the ERP specified values, provided that the leakage current resistance will increase to above 5 X $10^5$ ohms. The temperature criterion based on 5 X $10^5$ ohms would also apply to Pressurizer Level use in conjunction with RCS Subcooling for Safety Injection termination and reinitiation. If the ERP values for RCS subcooling are changed for Safety Injection termination, then a leakage current resistance of 1 X $10^5$ or greater would be acceptable for use.

Based on a review of Figure 1 and Figure 3.0-1, the instrument inaccuracy that exceeds the value that the operator can utilize with confidence occurs at a time when no operator action based on instrumentation in a harsh environment is required for the design basis events described above. For large and small LOCA, no mitigative or recovery operator actions are required based on instumentation in a harsh environment. For secondary breaks, Safety Injection termination (the required manual operator recovery action) will occur after the instrument accuracy returns to an acceptable value. Therefore, the operator limitation described in the previous paragraph will not prevent any necessary operator actions from being performed.

A review of the Reactor Protection System and Emergency Safeguards Features functions has determined that the significant functions required for harsh environment events (pressurizer pressure - Low SI and steam generator water level - Low-Low) are required only before 5 minutes after the event occurrence for pressurizer pressure - Low SI and 60 seconds for steam generator water level - Low-Low. This early time of use in the event should ensure that the function necessary will be performed before a significant error from leakage current develops.

TABLE

| PARAMETER | FUNCTION | TIME | DIVERSE PARAMETER | COMMENTS |
|---|---|---|---|---|
| 1. CTMT Sump Level | A. Identify LOCA | A. Short Term $<$ 20 min | (A-1) CTMT Radiation (BQlist)<br>(A-2) CTMT Pressure (NR or WR) | |
| | B. CTMT Recirculation | B. Long Term | RWST Level | Only verification – RWST level primary |
| | C. Critical Safety Function | C. Long Term | None | Beyond Design Basis for Flood |
| 2. CTMT Pressure | A. Identify LOCA | A. Short Term $\leq$ 20 min | (A-1) CTMT Radiation<br>(A-2) CTMT Sump | CTMT Pressure not affected by current leakage |
| | B. CTMT Integrity CSF | B. Long Term | NONE | |
| | C. Adverse CTMT for Instrumentation | E. Long Term | CTMT Temperature | |
| 3. Subcooling | A. SI Termination and Re-initiation | A. Long Term | (A-1) PZR level<br>(A-2) RCS Pressure (WR)<br>(A-3) PZR Pressure | Needed, Needs RCS Pressure + Temperature, margin available |
| | B. CSF Monitor | B. Long Term | NONE | |
| | C. RCP Trip | D. Short Term $\leq$ 15 min | CTMT Pressure | Trip on Adverse CTMT as backup |
| 4. WR RCS Pressure | A. SI termination, *trend only* | A. Long Term | (A-1) PZR level<br>(A-2) PZR Pressure, backup | |
| | B. RCS Subcooling | B. Long Term | PZR Pressure | Needed or backup |
| | C. CSF Integrity | C. Long Term | PZR Pressure | Not /DBA |
| | *D. RHR Pump Stop* | *D. Long Term* | | *Not required action* |
| 5. WRT (HOT) | A. RCS Subcooling | A. Long Term | (A-1) Core Exit TC<br>(A-2) WRT (Cold)<br>(A-3) SG Pressure | Needed, Unit 1 |

TABLE (continued)

| PARAMETER | FUNCTION | TIME | DIVERSE PARAMETER | COMMENTS |
|---|---|---|---|---|
| 6. WRT (Cold) | A. RCS Subcooling, Backup to WRT/(HOT) | A. Long Term | (A-1) Core Exit TC<br>(A-2) WRT (Hot)<br>(A-3) SG Pressure | Backup only |
| | B. Integrity CSF (PTS) | B. Long Term | NONE | NOT DBA, operator information |
| 7. WR SG Level | A. Backup to NR level | A. Long Term | (A-1) NR SG Level<br>(A-2) AFW Flow | Backup only |
| 8. NR SG Level | A. Verify heat sink for CSF, LOCA/STEAM Line Break | A. Long Term | (A-1) AFW Flow<br>(A-2) WR SG Level | Needed, or backup, one SG required for heat sink |
| 9. PZR Level | A. SI termination & re-initiation | A. Long Term | (A-2) RCS Subcooling<br>(A-2) WR RCS Pressure | No actions solely on PZR level |
| | | | (A-3) PZR Pressure | Above 1700 psig only |
| | B. CSF Inventory | B. Long Term | NONE | Only yellow path which is not required |
| 10. CETCs | A. Inadequate Core Cooling<br>B. RCS Subcooling | A. Long Term<br>B. Long Term | WR T(Hot)<br>(B-1) WR (THOT)<br>(B-2) WR T(COLD)<br>(B-3) SG Pressure | Not DBA<br>Unit 2 Only |
| *11. CTMT Radiation | A. Identify LOCA Backup only | A. Short Term < 20 min | (1-1) CTMT Sump<br>(1-2) CTMT Pressure | Backup only |
| | B. Adverse CTMT for Instrumentation | B. Long Term | Sample CTMT Atmosphere | |
| | C. CTMT Monitor for CSF | C. Long Term | Sample CTMT Atmosphere | To unisolate CTMT |
| *12. CTMT Temp. | A. Backup to CTMT Pressure for adverse CTMT instrumentation | A. Long Term | CTMT Pressure | Backup only |

*Not addressed in WCAP 11658, listed here only as potential backup instrument.

TABLE (continued)

ASSUMPTIONS

1) All Rx Trip/ESF in WCAP are SHORT TERM. They perform their function before they see a significant adverse CTMT. Even SG level for Rx trip and PZR Pressure SI perform their function before they see a significant adverse CTMT.

2) Short Term: 5 minutes Rx Trip/ESF
20 minutes other short term
Long Te - covers entire accident

3) 16 instruments are required (minimum #) for a DBA to reach Safe Shutdown.
   - 12 are in a harsh environment - see pages 1 thru 4
   - 4 are in a mild environment (not listed on Table)
     - AFW flow    - SG Pressure
     - RWST level    - CST level

4) Any other instruments required for post-accident monitoring (WCAP) are not required for DBA to reach Safe Shutdown.

NOV 24 '87 04:19 WEC-EAST 405A P.7

TERMINAL BLOCK IR VS TEMPERATURE

FROM SNL REPORT SAND 83-1617

FIG. A1-21 Pg. 210 PHASE 2 1ST PYT RAMP

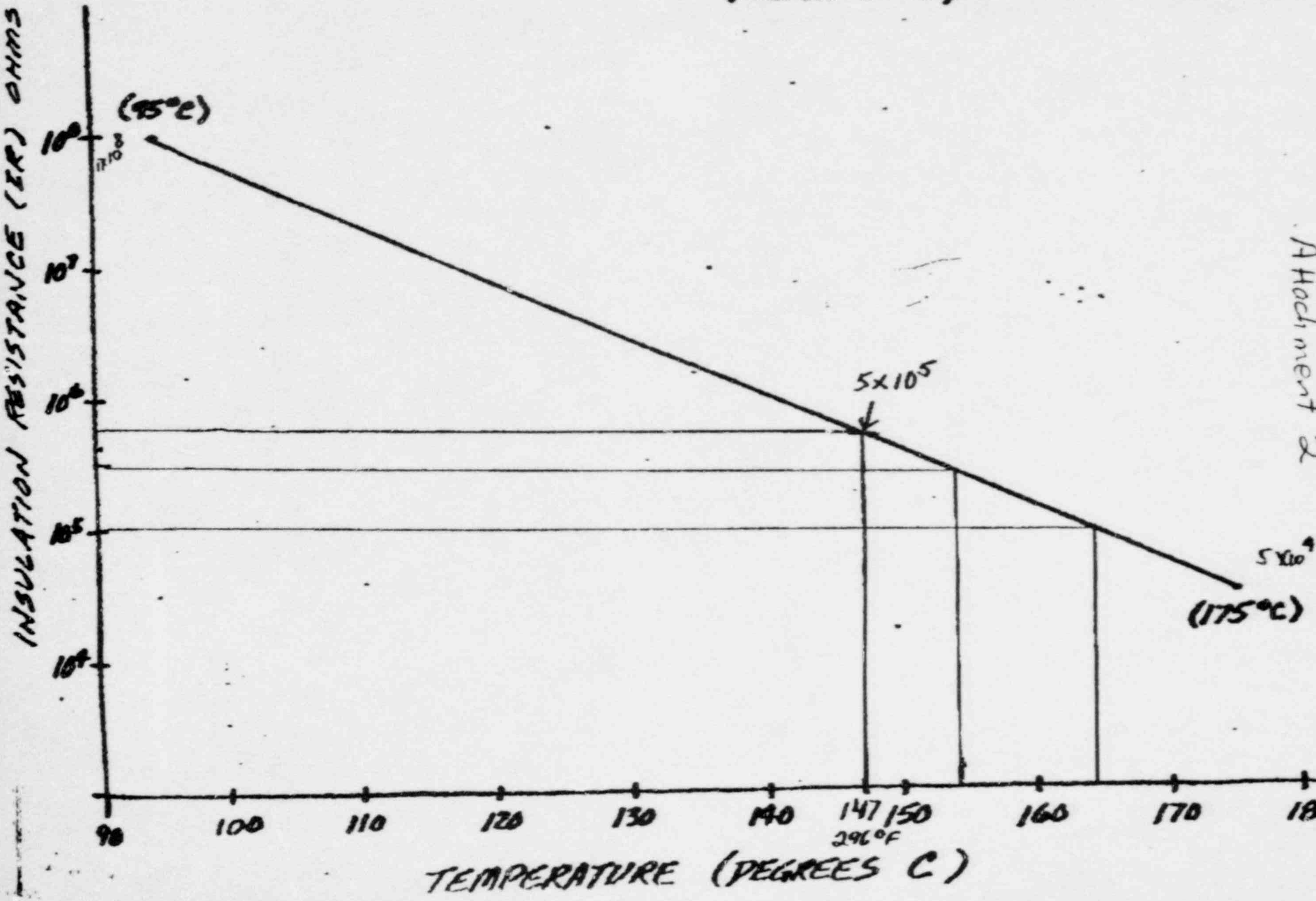



Figure 1
Attachment 2

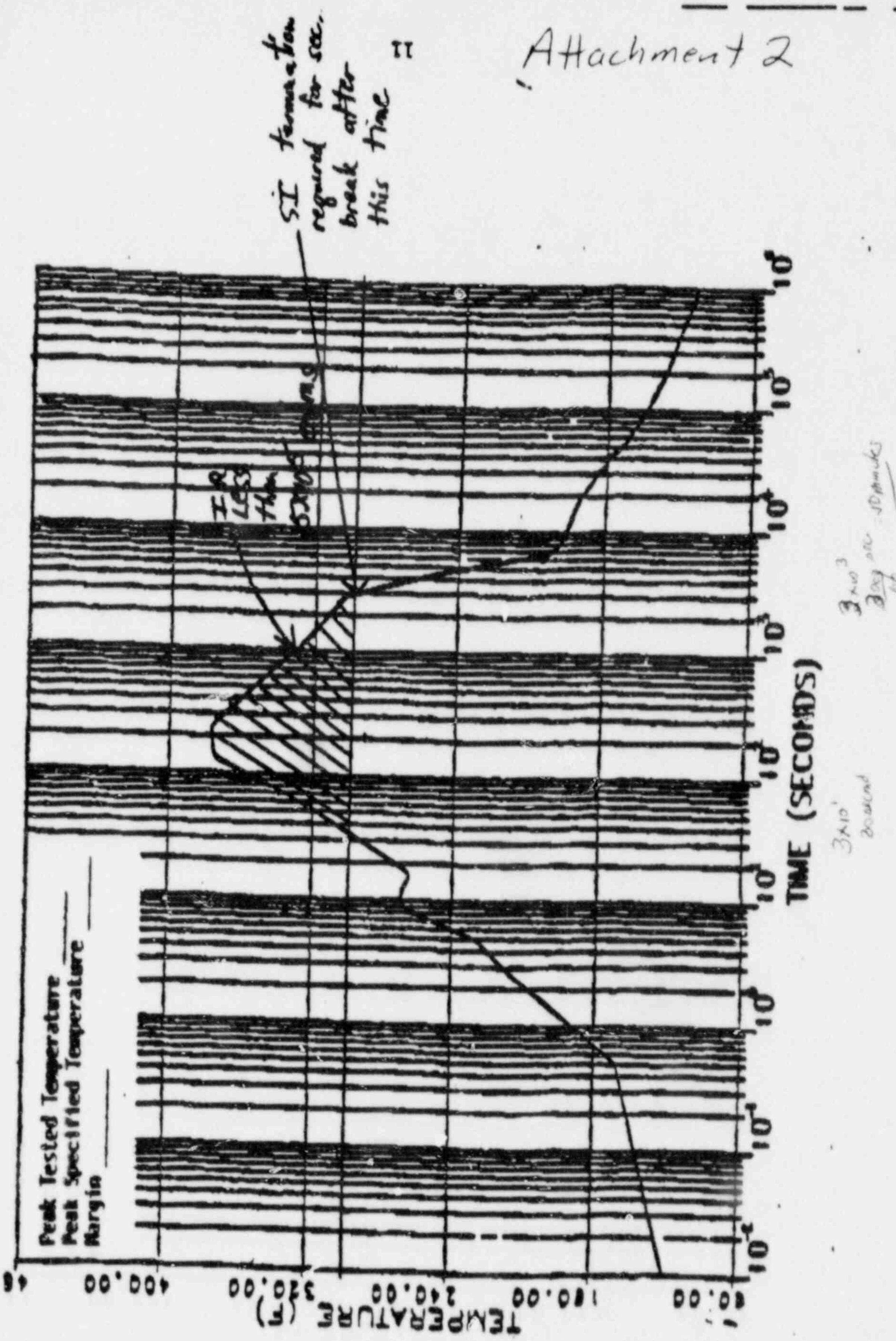
Attachment 2

FIGURE 3.0-1

This curve is based on FSAR Curves Figures 6.2-11, 6.2-13, and 6.2-40.

COMPOSITE LOCA/MSLB

# CONTAINMENT VAPOR TEMPERATURE ENVELOPE



Attachment 3
Figure 1

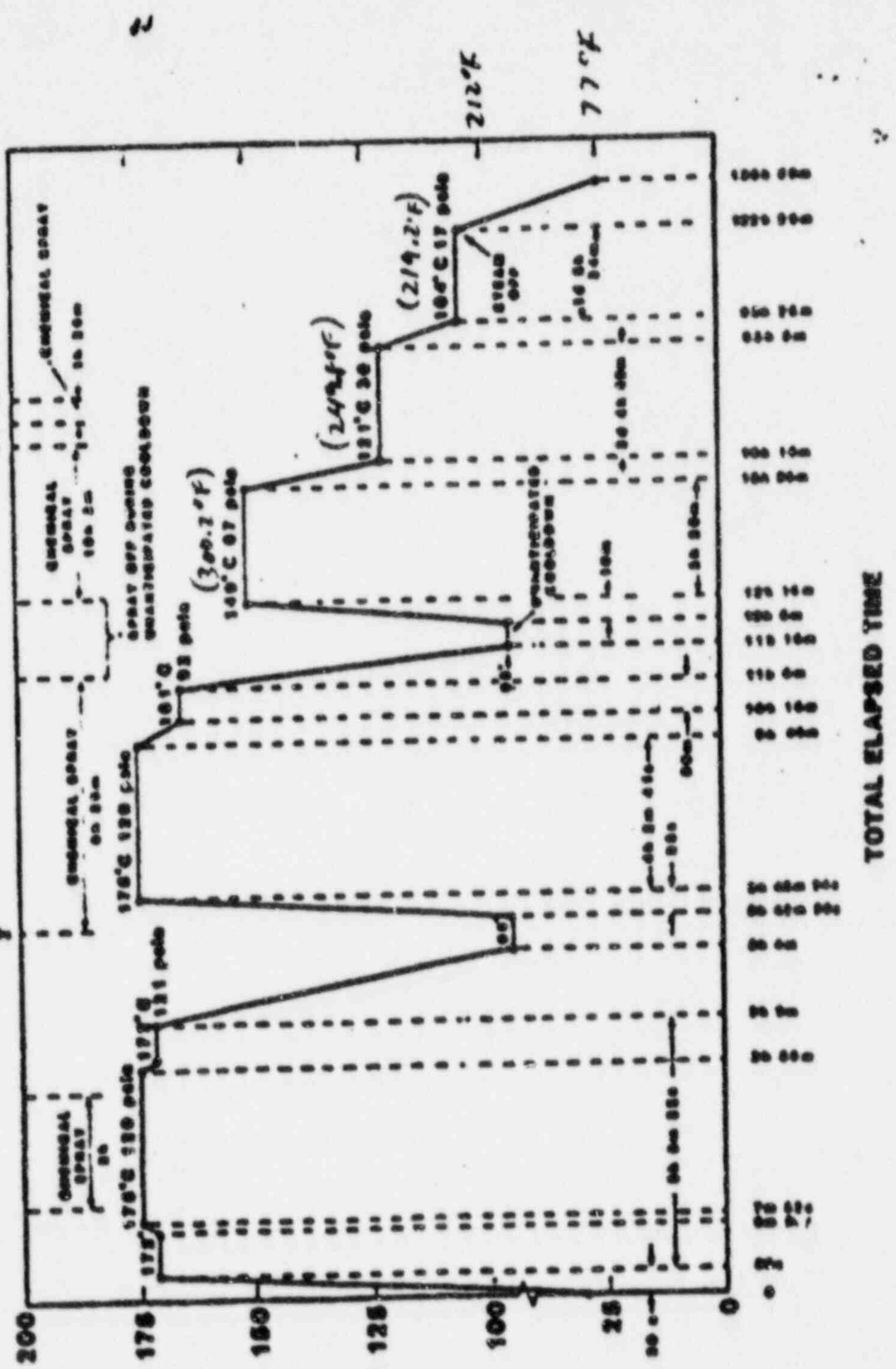



Figure 2

Phase II Environmental Temperature Profile

ENCLOSURE 3.(b)

# RAYCHEM/CHICO ENVIRONMENTAL SEAL QUALIFICATION

## NRC Proposed Violation:

The Chico Seal qualification package has not demonstrated that Raychem will bond to conduit.

## APCo Position:

The postulated failure mechanism discussed during the audit was chemical spray during a LOCA reacting with the zinc coating on the galvanized steel nipple to form a gray powder over the nipple. The result is a path for enough moisture to enter the limit switch between the Raychem and the degraded conduit causing the limit switch to fail. The following paragraphs describe in detail the Farley configuration and its configuration relative to the postulated failure described above. In summary, it should be noted that Chico A alone provides a pressure tight seal inside the pipe nipple which provides a pressure tight seal. To provide additional assurance that moisture will not enter the limit switch, three additional barriers have been applied to the FNP configuration. They are:

1) Raychem breakout boot

2) Keeper sleeve

3) Compression adapter clamp

The Raychem breakout kit used for the FNP application is environmentally qualified including thermal aging, irradiation, and LOCA testing (Reference Wyle Test Report No. 58442-2, dated 4/03/81). The Farley configuration uses a breakout at the end of pipe nipple. Since the breakout had been qualified previously, Farley conducted a test on the RAYCHEM/CHICO environmental seal configuration shown in Figure 1 for pressure and temperature conditions postulated during a LOCA (Reference Qualification Testing of Raychem Environmental Seals for Alabama Power Co., Joseph M. Farley Nuclear Plant, dated 12/30/81). The test did not include exposing the test specimen for chemical spray. The following paragraphs address the affect of chemical spray.

The environmental seals used with NAMCO EA-180 limit switches are composed of a Raychem WCSF breakout boot that has been shrunk onto a 1" pipe nipple attached to the limit switch (See Figure 1). The individual conductors connected to the switch pass through the breakout boot which forms a seal to the conductor insulation/jacket. To provide mechanical rigidity to the breakout boot, the nipple and the breakout boot are filled with Crouse-Hinds sealing compound (CHICO A) and allowed to cure. In addition to providing mechanical rigidity to the breakout boot crotch, the CHICO A provides an additional pressure tight barrier (seal) inside the pipe nipple which is environmentally qualified. CHICO A was qualified by test conducted by Southwest Research Institute (SWRI Project No. 03-4974-001) for use as drywell penetrations for Grand Gulf Nuclear Station. In addition, on the recommendation of Raychem, a keeper sleeve was installed over the breakout boot and the nipple to add rigidity to the boot, and to keep the boot in place during elevated accident temperatures when the adhesive softens.

APCo Position: (continued)

In the final assembly, an appleton compression adapter is clamped over the keeper sleeve to provide support for the flexible conduit, and it also mechanically clamps the keeper sleeve to the pipe nipple.

The zinc coating on the galvanized steel nipple may interact with the chemical spray during LOCA and form a gray powder over the nipple. However, the chemical spray does not react with the Raychem S1119 adhesive (Reference Wyle Test Report No. 58442-2, Section 3.4, Page 9 and Section 3.4.2). In addition to the duration of spray at Farley is only 87 minutes and the individual conductors will be effectively shielded from the spray.

Should there be a failure of the adhesive between the pipe nipple and the breakout boot, for whatever reason, the seal assembly would remain intact because of the keeper sleeve and the clamping action of the compression adapter. If it is postulated that the breakout boot, the keeper sleeve and the compression adapter clamp all fail, the internals of the NAMCO limit switch will still be protected by the approximately 3 inch long CHICO A seal.

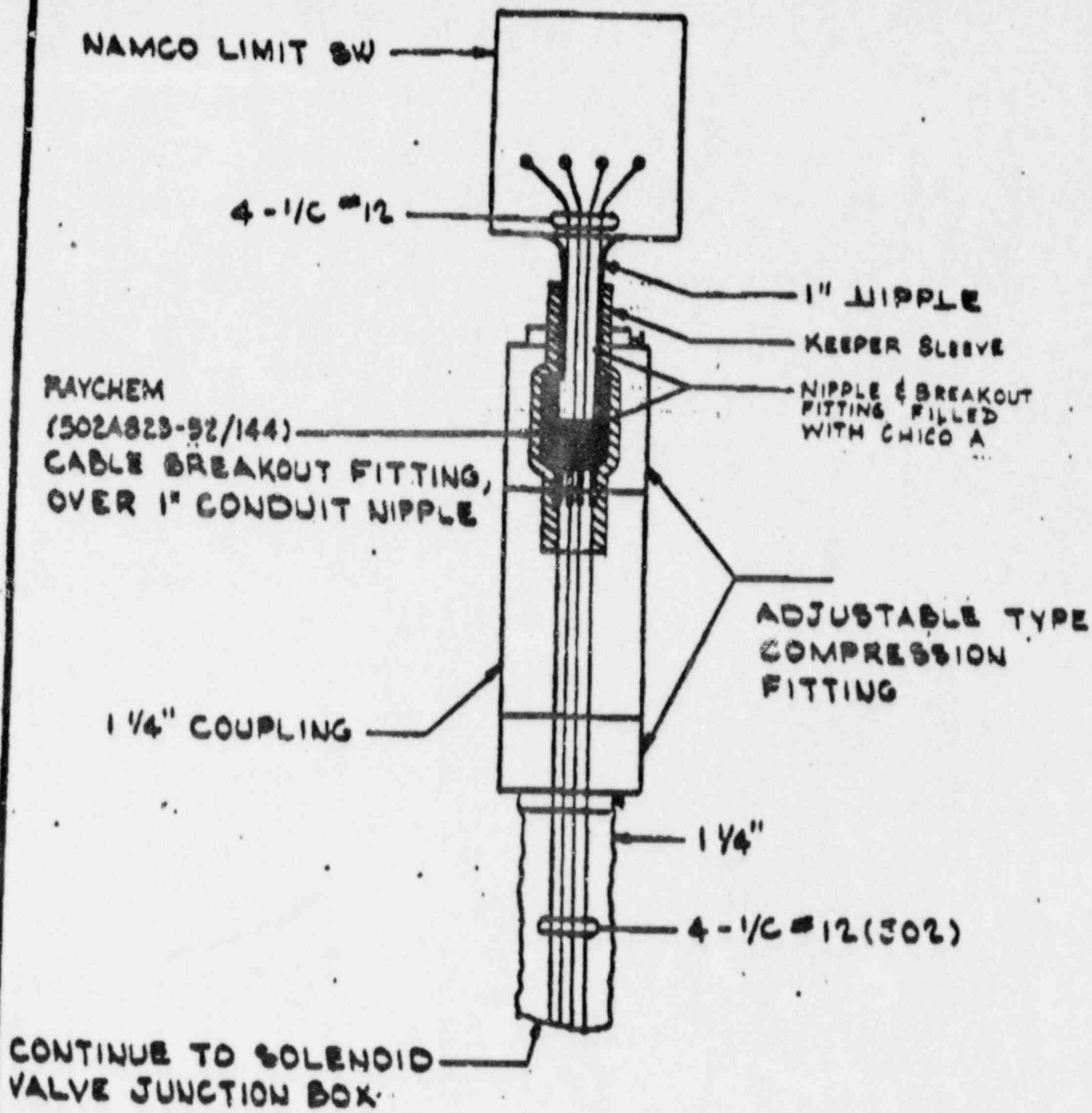

NAMCO LIMIT SW
4-1/C #12
1" NIPPLE
KEEPER SLEEVE
RAYCHEM
(502A823-52/144)
CABLE BREAKOUT FITTING,
OVER 1" CONDUIT NIPPLE
NIPPLE & BREAKOUT FITTING FILLED WITH CHICO A
ADJUSTABLE TYPE COMPRESSION FITTING
1 1/4" COUPLING
1 1/4"
4-1/C #12(302)
CONTINUE TO SOLENOID VALVE JUNCTION BOX

FIGURE 1