ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9011010076 DOC.DATE: 90/10/23 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

AUTH.NAME	AUTHOR AFFILIATION
ALEXICH,M.P.	Indiana Michigan Power Co.
RECIP.NAME	RECIPIENT AFFILIATION
MURLEY,T.E.	Document Control Branch (Document Control Desk)

SEE REPORTS

SUBJECT: Summarizes 901004 meeting w/NRC in Rockville,MD,to discuss SSFI rept.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 61
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-1 LA	1	1	PD3-1 PD	1	1
	COLBURN,T.	2	2			
INTERNAL:	NRR/DET/ECMB 9H	1	1	NRR/DOEA/OTSB11	1	1
	NRR/DST 8E2	1	1	NRR/DST/SELB 8D	1	1
	NRR/DST/SICB 7E	1	1	NRR/DST/SRXB 8E	1	1
	NUDOCS-ABSTRACT	1	1	OC/LFMB	1	0
	OGC/HDS1	1	0	~~REG FILE 01~~	1	1
	RES/DSIR/EIB	1	1			
EXTERNAL:	NRC PDR	1	1	NSIC	1	1

R
I
D
S
/
A
D
D
S

R
I
D
S
/
A
D
D
S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 17 ENCL 15

Indiana Michigan
Power Company
P.O. Box 16631
Columbus, OH 43216

INDIANA
MICHIGAN
POWER

AEP:NRC:1138

Donald C. Cook Nuclear Plant Units 1 and 2
License Nos. 50-315 and 50-316
Docket Nos. DPR-58 and DPR-74
SUMMARY OF OCTOBER 4, 1990, MEETING TO DISCUSS SSFI REPORT

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

Attn: T. E. Murley

October 23, 1990

Dear Dr. Murley:

Pursuant to the direction of the NRC staff, the purpose of this letter is to summarize a meeting held at the NRC offices in White Flint, Maryland, on October 4, 1990 to discuss the SSFI report.

The meeting was held for the purpose of discussing the inspection report from the recently conducted safety system functional inspection (SSFI) of the Donald C. Cook Nuclear Plant essential service water systems. In particular, the meeting focused on issues in the inspection report related to design verification. The major finding of the SSFI was that design verification efforts for the Cook Nuclear Plant were not being properly implemented. The problems with design verification were linked, in the inspection report, to several perceived design deficiencies. Although we recognize that the inspection report raised valid concerns with the implementation of our design verification procedures, we believe that the tone of the report inappropriately characterized the depth of our design verification problems and the quality of our design work.

Attachment 1 contains a summary of our discussion on design verification. It includes details on the specific SSFI findings, a historical review of the evolution of our design control programs, and corrective actions we have taken to correct the deficiencies noted in the inspection report.

Attachment 2 contains a summary of specific technical issues from the inspection report that were discussed during the meeting.

A001
11

During the discussions, we presented our design and technical specification testing philosophy and provided the reasons we believe our design and testing are adequate. Moreover, we attempted to demonstrate that these issues represented technical differences of opinion, rather than deficient designs or examples of problems caused by inadequate design verification.

Attachment 3 contains additional information on issues that were raised in the inspection report but were not discussed during the October 4 meeting. This information is submitted, at your staff's invitation, in order to clarify our position on items that time did not permit us to discuss. Attachment 4 contains a copy of the overheads we presented at the October 4 meeting.

This document has been prepared following Corporation procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

ldp

Attachments

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
J. R. Padgett
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

ATTACHMENT 1 TO AEP:NRC:1138

SUMMARY OF DESIGN VERIFICATION ISSUES
DISCUSSED DURING OCTOBER 4, 1990 MEETING

The Safety System Functional Inspection (SSFI) Report for the Cook Nuclear Plant essential service water (ESW) system drew several conclusions in the area of design verification. While the findings from which the team drew its conclusions are factual, we contend that the conclusions inappropriately characterize the quality of design activities and the depth of problems in the area of design verification. In order to support this contention, this attachment considers the SSFI report findings on design verification in several steps.

First, a summary of the SSFI design verification findings is presented, including a discussion of the implications of these findings. Second, the topic of design control and design verification is reviewed from a historical perspective. This review considers original plant design practices; design control and verification methods for early design changes and the NRC's acceptance of these practices; and some of the more recent findings in this area, prior to the SSFI, including actions taken to correct these problems. Third, conclusions are drawn for the design verification topic based on the material presented. Lastly, our actions to respond to the SSFI issues are discussed.

1) <u>Design Verification Findings</u>

The report findings and conclusions for the area of design control and design verification are contained in Section 2.1.1 of the inspection report. Our review of these topics led to the conclusion that the findings could be categorized into three topics:

A) New procedural requirements,

B) Independent design verifications, and

C) Method of independent design verification for original plant design activities.

These three topics are addressed individually.

A) <u>New Procedural Requirements</u>

This topic was summarized in the following statement from the inspection report:

> Calculations performed after the new program was implemented in both electrical and mechanical designs either did not receive an independent design verification or received an improperly performed verification. In addition, verification of many post-1988 calculations were accomplished by the verifier stating not applicable (N/A) for each of the verification items on the independent verification forms....

This observation was substantiated during the inspection period by examples of lack of attention to detail in performing design verifications. While the team did conclude that our procedures adequately put forth the requirements of ANSI N45.2.11, success was not entirely achieved in implementing these procedures. On a positive note, the team did not construe any of these findings as causing deficient design outputs. However, we acknowledge that significant improvements must be made in this area. A discussion of corrective and preventive actions on this subject will be presented at the end of this attachment.

B) <u>Independent Design Verification</u>

The second category of findings involved calculations which the team cited as not receiving independent design verifications, but which did receive applicable reviews. The report gave three example calculations in this category:

1) Essential service water pump room temperature (Calculation No. DCCHV12ES), and

2) Essential service water flow requirements during a fire (Calculation No. HXP890720AF),

3) Essential service water pump potential runout (Calculation No. HXP900613).

Although these calculations were characterized as not receiving independent design verifications, the first two calculations were verified to the requirements in effect at the time they were performed. Mechanical Engineering Division Procedure No. 8 (MED-8), which was in effect until it was replaced with a Nuclear Engineering Department calculation procedure in August of 1989, allowed verification by a number of methods which were annotated on the calculation cover sheet. The cover sheet from the ESW pump room temperature calculation showed that Revision 0 of the calculation was verified by method 1 of MED-8, which constituted a check of the input data, method and numerical accuracy. The team's finding was based on the absence of an ANSI-type verification checklist, which is now required by the current Nuclear Engineering Department calculation procedure. This particular calculation was revised during the inspection period and included a calculation verification checklist, according to current requirements. Thus the inspection team noted that "the calculation was subsequently verified." This situation was virtually identical for the second example calculation, "ESW Flow Requirements During a Fire."

The third calculation was one done specifically to respond to an inspector's question during the SSFI and, although the verification checklist was completed by an independent reviewer, the section manager failed to sign it. However, the manager did sign the calculation cover sheet, noting his approval shortly after the calculation and verification were performed. It is important to note that the verification checklist is attached to the calculation, and the manager's signature on the cover sheet represents approval of the entire calculation package. Nuclear Engineering Department Procedure (NEP) No. 6.4 recognizes this fact and allows the manager to sign the calculation cover sheet as a minimum. In any event, the independent verification was performed, and the manager did approve the overall calculation which contained the verification checklist as an attachment. Following the SSFI, a procedure change was initiated for NEP 6.4 to drop the section manager's approval on the calculation verification checklist, since approval is noted on the calculation cover sheet.

Significant points from these three examples are:

1) None of these examples involved a technical problem with the calculation reviewed. All example findings were directed at design verification practices.

2) Procedural requirements have been augmented to always require completion of a design verification checklist for current calculations. Older calculations which are revised in the future will be verified using the checklist.

C) <u>Method of Independent Design Verification for Original Plant Design Activities</u>

The final category of findings was labeled "lack of independent design verifications or improperly completed verifications for original plant design calculations, test reports and drawings, and design documents resulting from design changes and design activities subsequent to plant licensing," in the inspection report.

This finding was directed at our original design practices as well as design changes that occurred from licensing up until the present. Several examples were cited on Page 3 of the inspection report which were noted as examples where inadequate design verification contributed to design

deficiencies. While we agree that the design verifications for these early design activities may not have been completely documented in the literal spirit of ANSI N45.2.11, the examples used are more technical disagreements between inspector and licensee than instances of "poor design" which resulted from inadequate design verifications.

2) Historical Perspective

Original design practices for the Cook Nuclear Plant were the subject of considerable attention in the 1983-1984 timeframe. During and following a 1983 inspection (Report Nos. 50-315/83-18 and 50-316/83-19), there was significant dialogue concerning original design control practices and early design changes. In response to these events, we revised our corporate level and organization-specific procedures to more fully implement the design verification requirements of ANSI N45.2.11-1974.

Additionally, we performed a historical review of a sample of early design changes to ensure that design control and verification practices produced acceptable results. This sample of design changes was subjected to the design verification criteria set forth in ANSI N45.2.11-1974, and we concluded that our early design practices and verification methods were effective.

Finally, in 1984 we submitted a description of original design control practices for the Cook Nuclear Plant to the NRC (AEP:NRC:0858B). In developing this description, we concluded that original design practices were consistent with good engineering practice. Although a large portion of the original design was performed prior to the publication of ANSI N45.2.11-1974, the design concept implemented included a cognizant engineer and a number of reviews. Though different in form and ease of auditability from the requirements of ANSI, we concluded that the original design concept was commensurate with the design verification requirements described in 10CFR50, Appendix B, Criterion III, and ANSI N45.2.11 in ensuring an effective design process.

During a 1984 NRC inspection which considered these issues and reviewed numerous procedures and activities (Inspection Report Nos. 50-315/84-16 and 50-316/84-18), inspectors concluded that top tier and implementing procedures properly included requirements for design verification per ANSI N45.2.11-1974.

This point is significant in that these 1984 conclusions were based on a review of some of the same procedures that resulted in the verifications to which the SSFI team took exception. For example, the 1984 inspection team reviewed MED-8 and noted that the procedure "included new provisions...for calculation review and independence of the reviewer." As noted previously, the SSFI team characterized some of the calculations verified per this procedure as examples of calculations which did not receive an independent design verification, which were subsequently verified. The "subsequent verification" included the use of a formal ANSI-type design review checklist, which is now required by current procedures.

The 1984 inspection report also concluded that:

> The design and verification activities conducted during the original plant design development were consistent with generally accepted engineering practices for the time they were accomplished. Review and verification activities were performed in a manner considered to be good engineering practice.

Therefore, the SSFI finding that independent, formal design verifications in the spirit of ANSI N45.2.11 were not performed for some original design activities, while accurate, has been acknowledged, and early methods were found to be acceptable.

Following these 1984 events, which resulted in favorable reviews of our design verification methods, we had no indication of design verification problems until 1987, when WESTEC was contracted to perform a SSFI of the Unit 1 auxiliary feedwater system. In its SSFI report, WESTEC noted apparent problems with traceability of design inputs, particularly for early calculations, lack of stand alone documentation packages, depth of verification practices, and information flow across interdisciplinary boundaries.

In addition to addressing the symptoms noted in the WESTEC report, in late 1988 we undertook a significant reorganization within the corporate engineering support groups for the Cook Nuclear Plant. In particular, the electrical and mechanical engineering divisions were consolidated into a single Nuclear Engineering Department. Part of this reorganization included the development of a single set of department level procedures. The procedure development effort critically considered the ANSI design control requirements, past inspection findings in the area of design control, and the "best" practices of the former

electrical and mechanical engineering department procedures. These procedures were those characterized by the SSFI inspection team as properly putting forth the requirements of ANSI N45.2.11-1974.

During an NRC inspection that spanned from late 1988 into 1989, we received additional findings in design control, particularly in the area of piping design. Specific weaknesses were in the area of quality of documentation for design changes, the use of unsubstantiated judgements and adequacy of verification practices.

In response to these findings, a reorganization was undertaken within the Design Department in late 1989, which resulted in the formation of a Nuclear Design Group. This new group is responsible for all design support activities for the Cook Nuclear Plant including electrical, mechanical, structural/analytical, and site design. As with the Nuclear Engineering Department formation the previous year, a key facet of this reorganization was the development of new procedures that included pertinent requirements and recent issues.

3) <u>Conclusions</u>

As previously discussed, significant effort has been expended to develop procedures which set forth the ANSI design control requirements. These procedures, which the SSFI team considered acceptable, are those developed as part of recent reorganizations in both the engineering and design departments. However, we recognize that we have not been entirely successful in implementing these new procedural requirements. Ongoing and planned future actions to address this issue will be discussed later. Nevertheless, several conclusions can still be drawn from the discussions presented.

While improvement must be made in performance and documentation of design verifications to our new procedural requirements, it is important to note that some of the examples of past activities cited as not having design verifications did receive applicable reviews according to methods previously accepted by the NRC. Additionally, none of these examples were termed "safety-significant" by the inspection team.

Additionally, it should be noted that several of the findings were directed at original design practices and the verifications inherent in our practices. These methods were found to be acceptable during an earlier inspection.

4) Actions to Respond to SSFI Design Verification Issues

A number of actions were taken following the SSFI because of the noted deficiencies in procedure implementation and other actions are planned.

Shortly following the SSFI, in August 1990, a training session was administered to familiarize personnel with the design verification findings noted during the SSFI. Specific deficiencies noted by the inspection team were used as examples for this training. This training was completed prior to receipt of the SSFI inspection report.

Upon receipt of the SSFI report, senior management directed that the Nuclear Engineering and Design Departments develop action plans to respond to the SSFI findings. These action plans have been developed and include the following:

1) A historical review of a sample of design outputs that were originated under our new procedures. This review is intended to ensure that these outputs were properly completed according to these procedural requirements. This review began in the fourth quarter of 1990.

2) Formation of quality review teams (QRTs) to review a sample of future design output documents. This review is to ensure that outputs are technically adequate and procedurally correct. QRT activity is commencing in the fourth quarter of 1990.

3) Development of subject training in design control, to ensure that personnel fully understand procedural requirements for assigned tasks. Development of lesson plans is underway, and training will commence early next year.

Items 1 and 3 will be completed by the end of 1991.

In addition to these steps, on September 21, 1990, Mr. D. H. Williams, Jr., our Senior Executive Vice President - Engineering and Construction, reissued a previous policy statement on the requirement for strict procedural adherence. The directive that effected the reissue emphasized that employees' performance would be rated based on how effectively and completely the employee complies with procedures governing his or her work.

ATTACHMENT 2 TO AEP:NRC:1138

SUMMARY OF TECHNICAL ISSUES DISCUSSED
DURING OCTOBER 4, 1990 MEETING

This attachment provides a summary of six technical issues that were discussed in some depth during the October 4, 1990 meeting. The six issues, and the references to the appropriate report sections are provided below.

Issue	Reference
1) ESW Thermal Relief Valves	Section 2.1.3, pg. 5; Section 2.1.1, pg. 3
2) Cable Sizing	Section 2.1.2, pg. 5; Section 2.1.1, pg. 3
3) Breaker Ratings	Section 2.1.2, pg. 5; Section 2.1.1, pg. 3
4) Inverter Rating	Section 2.1.2, pg. 4; Section 2.1.1, pg. 3; Unresolved item 90-201-03, pg. A-3
5) Diesel Generator Sizing	Section 2.1.2, pg. 4
6) Battery Testing	Section 2.5.2, pg. 19; Unresolved item 90-201-01, pg. A-1; Unresolved item 90-201-08, pg. A-9

Each of these issues is discussed below.

ISSUE NO. 1: Essential Service Water System Thermal Relief Valves

Inspection Report Text (pg. 5)

"The design documentation indicated that the thermal relief valves on the component cooling water heat exchangers, diesel generator heat exchangers, and containment spray heat exchangers had been set at 150 psig while the piping system design pressure was 105 psig. This was inconsistent with the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code), Section III, Article ND7000, which required that the set pressure of the relief valves be at, or lower than, the design pressure. The licensee stated that the relief valve setpoints would be lowered to 105 psig."

Response

The configuration traces back to the original ESW system design, which has a 105 psig design pressure, corresponding to the shutoff pressure of the ESW pumps. When the system equipment was specified, it was common practice to specify components with design pressures higher than the system design pressure, which is conservative. Typically a 150 psig component design pressure was used.

The code of construction for the service water side of the ESW system heat exchangers, ASME Section VIII - 1968, requires that a pressure relieving device be provided which prevents the pressure from rising more than 10 percent above the maximum allowable working pressure (assumed to be the same as the design pressure, as allowed by the code). Since the only source of potential overpressure above pump shutoff head is thermal expansion when the heat exchanger is isolated, 150 psig relief (or sentinel) valves were provided, corresponding to the component design pressure.

During an internal QA review of an ASME repair plan for the ESW system, the question of code requirements and boundaries arose. The 1974 version of ASME Section XI, which Cook Nuclear Plant was committed to for the first 10-year inspection interval, called for a post-repair hydrostatic test at a pressure of 1.10 times design pressure, which was taken as system design pressure (105 psig). The 1983 version of ASME Section XI, which has been adopted for the second 10-year inspection interval, calls for post repair hydrostatic tests at 1.10 times the lowest safety valve setting. Application of this criteria to the ESW system would have required unnecessarily testing the non-isolable portions of piping near components to 1.1 times 150 psig, or 165 psig. Since the 16" standard weight ESW piping has a maximum working pressure of 625 psig, there would have been no problem with doing this. However, it would have resulted in changes to the minimum wall thickness

requirements for the erosion/corrosion program and would also have required hydrostatically testing the system at two pressures. Therefore, as a matter of convenience, it was decided to lower the set pressure of the sentinel valves to the piping system design pressure of 105 psig to provide literal compliance with the 1983 version of ASME Section XI. A problem report was written to ensure a thorough review of this issue. The investigation of this problem report resulted in a planned design change to lower the safety valve settings for ESW system components to 105 psig.

The issue was characterized within the SSFI report as an example of an inadequate design verification leading to a deficient design. We contend that this was a mischaracterization since the original design was acceptable; the plan to lower the safety valve setting is a matter of convenience for hydrostatic testing purposes.

ISSUE NO. 2: Cable Sizing

Inspection Report Text (pg. 5)

"Cable sizing calculations did not verify the capability of the ESW pump feeder cable to withstand effects of short circuit currents until these currents were interrupted by the upstream breaker.... This appeared to be a generic design problem for all of the safety-related cables. However, the team verified that a single failure due to a short circuit would result in the loss of a single piece of equipment, and that redundant equipment would not be damaged. Therefore, no immediate safety concern existed. However, the licensee agreed to review this issue."

Response

This consideration, the capability of the pump feeder cable to withstand short circuit currents, was not a design basis consideration for the Cook Nuclear Plant. This type of design consideration requires possible oversizing of cable conductors such that a worst-case fault current will not cause the conductor surface temperature to exceed the published emergency temperature rating, with resultant insulation damage. For the Cook Nuclear Plant, our design basis required sizing per the IPCEA ampacity tables.

The consideration of cable damage due to maximum fault current is considered to be an economic concern, not a safety concern. As stated in the inspection report, protection for the bus is provided by the upstream breaker. The design of the buses is such that no single failure/fault can take out redundant trains of safety-related equipment. Additionally, we note that the type of fault necessary to cause cable damage is considered to be highly unlikely. For the ESW feeder cable discussed in the inspection report, a three-phase fault would be required, at approximately mid-cable. Since individually shielded, grounded, phase cables are used, a three-phase fault at a location other than the terminal ends is highly unlikely. For a fault at the terminal end, the impedance of the cable reduces the fault current to a level that does not damage the insulation. At the Cook Nuclear Plant, we do not allow repair to a faulted power cable. If a cable is damaged, the entire cable is replaced. Also, if an end device (e.g., a motor) is faulted, the cable supplying the device would be tested for damage prior to returning the equipment to service.

ISSUE NO. 3: Breaker Ratings

Inspection Report Text

"The [short circuit] calculations indicated that the breakers for the safety-related and nonsafety-related 4KV buses were undersized. For the 4160-V safety-related buses, the maximum calculated short circuit duty was found to be about 69,000 amps, but the installed breakers were rated for a maximum of 60,000 amps....The team verified that the loss of a breaker would not affect redundant equipment. Therefore, no immediate safety concern existed. However, the licensee agreed to review this issue."

Response

This issue had been self-identified in 1988. At that time, both technical and licensing basis reviews were performed to provide assurance that the design was adequate and consistent with our licensing basis.

The technical review determined that this was a very low probability concern. In order for the fault current to exceed the breaker rating, a bolted three-phase fault must occur at or near the breaker in conjunction with maximum DC offset and the system in an atypical configuration. (By atypical, it is meant that the fault would have to occur while the unit is powered by the reserve (offsite) source, rather than the generator auxiliary transformers.) A bolted three-phase fault at or near the breaker is considered highly unlikely. The switchgear is enclosed in metal cabinets which are not accessed with the equipment energized. The power cables are individually shielded, grounded, phase cables, making the likely failure a phase-to-ground fault. The Cook Nuclear Plant electrical distribution system is a resistance ground system which limits the phase-to-ground fault current to approximately 2,000 amps.

The safety-related electrical buses are designed such that no single fault can affect redundant trains of safety equipment. Besides being separated electrically, the electrical equipment trains are physically separated such that if a breaker were to be physically damaged, the other train would not be impacted by the damage.

In conclusion, the 1988 review reaffirmed the adequacy of the original design from both a technical and a licensing perspective.

ISSUE NO. 4: Inverter Rating

Inspection Report Text (pg. A-3)

"Because the end-of-life (EOL) voltage at the station battery terminals was calculated to be 210 volts, inverter terminal voltage during battery EOL would always be less than 210 volts due to feeder voltage drop. Therefore, the inverter would not be operational. This condition could result in an inadequate power supply to plant instrumentation during a loss of ac power supply. During the inspection the licensee initiated actions to requalify these inverters for a minimum input voltage of 200 volts dc."

Response

Per Technical Specification Table 4.8-2, the inverters are only required to function for three hours in an emergency. Under the newer station blackout requirements, the inverters are only required for four hours. At the end of four hours, the battery terminal voltage is calculated to be 226 volts, with the corresponding voltage at the inverter being 220 volts, which is above the inverter's specified minimum rating of 210 volts. We believe this adequately addresses the NRC's concerns.

ISSUE NO. 5: Diesel Generator Sizing

Inspection Report Text (pg. 4)

"The initial 1CD EDG sizing calculation was unconservative because low service factors were assigned to cyclic loads such as the boric acid tank heaters and the boric acid heat trace and load center transformer efficiencies were omitted. The licensee performed another calculation for EDG sizing during this inspection which accounted for transformer losses, cyclic loads and increased loading of approximately 48 kW as a result of an undetected ground fault. The result of this calculation indicated that the load for the 1CD EDG was approximately 3590 kW, thus exceeding the continuous rating of 3500 kW, but below its 2,000 hour rating of 3650 kW during LOCA and LOOP conditions. Because the latest calculation was conservative in its use of assumptions, the team agreed that the EDG would meet its maximum load demands. However, the team noted that the licensee had not performed dynamic analysis to determine actual fluctuations of EDG loads. The licensee stated that EDG dynamic analysis would be performed."

Response

Contrary to what is stated in the inspection report, a dynamic analysis of the emergency diesel generator (EDG) had been completed in 1972.

In the sizing calculation discussed in the inspection report, a 50% diversification factor was applied to loads such as heat tracing, EDG room heaters, EDG starting air compressors, and the auxiliary building elevator. The 50% diversification factor is considered to be a conservative assumption, given the nature of these loads. For example, the thermostatically controlled EDG room heaters are not expected to be in service while the EDG is running because of the large quantity of heat generated by the EDG. The calculation employed other conservative assumptions, such as assuming that all 4kV motors were at maximum loading and that motors were 90% efficient. (The motors are specified to have at least 93% efficiency.)

As discussed in the inspection report, even with elimination of the 50% diversification factor, the 1 CD EDG is barely over its continuous duty rating and well within its 2,000 hour rating of 3650 kW. The calculated loads on the other EDGs are below their continuous duty rating. All of the EDGs have previously been tested at their 2,000 hour rating. For these reasons, we believe the EDGs are being applied within their ratings and do not believe a new dynamic analysis is warranted.

ISSUE NO. 6: Battery Testing

Inspection Report Text (pg. 19)

"Battery sizing documents prepared by the utility engineers during 1984 and 1985, in preparation for the purchase of replacement batteries, showed that the battery load profile developed by the engineers exceeded the identified load profile of the technical specification by 35 to 65 percent. Following the installation of the new batteries in 1986, the licensee failed to incorporate the new battery capacity and test profile into existing test procedures (2MHP4030.STP.034, 2MHP4030.STP.022, and 1MHP4030.STP.044), and the 18-month surveillance procedures for 2AB, 2CD, and 1AB battery emergency load discharge and battery charger tests. Therefore, the technicians continued to test the new station batteries to a load profile that was up to 65 percent below the calculated emergency loads. The licensee, by not incorporating the 1984 battery load profile into the applicable battery capacity test, did not meet the requirements of 10 CFR Part 50, Appendix B, Criterion XI. This criterion required a test program to demonstrate that components will perform satisfactorily in service in accordance with written test procedures that incorporate the requirements and acceptance limits contained in applicable design documents."

Response

The sizing calculations were performed in order to procure new station batteries. The calculations were performed in accordance with the manufacturer's battery sizing worksheets. The worksheets are similar to those of IEEE 485 in that they both require temperature correction, design margin, and aging factors. Using these sizing worksheets, an ampere hour battery of sufficient size to meet the duty cycle was chosen. These calculations indicate that the batteries can supply the emergency loads.

Capacity and performance tests were conducted after installation and are routinely performed on station batteries every 60 months. This demonstrates the battery capacity and detects any changes in the capacity determined by the original acceptance tests.

Prior to May 1989, the Technical Specifications did not require a service test as defined in IEEE-450. Emergency loads were identified and durations specified during the 18-month surveillance test as specified in the Technical Specifications.

Following discussions with INPO, we reviewed our battery testing methods and determined that a service test more in line with IEEE-450 could be performed using a simulated load established by a computerized battery load testing device. Because the Technical Specifications did not recognize the use of simulated loads, a

Technical Specification change was necessary to use the simulated loads. The required Technical Specification change was received from the NRC in May 1989. This was during the Unit 1 outage, in time to permit only the testing of the 1CD battery with the new method. The 1AB battery had been tested earlier in the outage to the previous technical specification requirements.

As of this date, both Unit 2 batteries have been tested using the computerized test method. The 1AB battery will be tested during the upcoming Unit 1 outage. The 1AB load profile is similar to those of the other batteries, and is in fact enveloped by the 1CD profile. Since the batteries were procured from the same supplier seven years ago and have been subjected to the same maintenance practices, we do not expect to encounter problems when testing the 1AB battery.

In conclusion, testing was performed in accordance with Technical Specification requirements in-place at the time of the testing. We agree that the IEEE-450 method of testing using a computerized load testing device is more rigorous and have taken the appropriate engineering and licensing steps to incorporate the service testing into our surveillances. We are confident, based on the facts set forth in the above discussion, that the batteries will meet their intended design function.

ATTACHMENT 3 TO AEP:NRC:1138

CLARIFICATION OF MISCELLANEOUS ISSUES
FROM THE SSFI INSPECTION REPORT

The following provides clarification of our position on several miscellaneous items discussed in the inspection report.

ISSUE NO. 1: I&C Setpoint Methodology

(Ref. Inspection Report Section 2.1.4, pg. 8)

Inspection Report Text (pg. 8)

"The licensee stated that the [I&C] setpoint calculation procedure will be improved using the guidance of Regulatory Guide 1.105, 'Instrument Setpoints for Safety Related Systems,' Revision 2, February 1986."

Clarification

We did not commit to improve our setpoint procedures using the guidance of Rev. 2 of Regulatory Guide 1.105. Rather, our statement, as provided in a position paper provided to the inspection team, was as follows:

"We are 'evolving' into using ISA methodology that is currently in draft version known as part of SP 67.04 Part II. It is our intention at this time to use this methodology on a consistent basis once the ISA recommended practice is finalized, approved, and issued. We recognize that this recommended practice is also expected to be endorsed by a revision to Reg. Guide 1.105."

ISSUE NO. 2: Vendor Manual Review

(Ref. Inspection Report Section 2.3.1, pg. 12 and Unresolved item 90-201-05, pg. A-5)

Inspection Report Text (pg. A-5)

"Procedure 12PMP2030.VICS.001, "Control of Vendor Documents," Revision 2, established a vendor information control system (VICS). Section 5.0 of that procedure required that all vendor information including bulletins, letters, vendor manuals or revisions be processed and controlled to ensure their proper availability and use under the licensee's document control system. The licensee failed to identify and include a March 19, 1984 vendor letter regarding the diesel bleeddown test acceptance criteria in the vendor file for the emergency diesel generators."

Clarification

The VICS system for the Cook Nuclear Plant was not established until 1985. Thus, VICS was not in use at the time the March 19, 1984 letter was received. Although not processed through VICS, the information in the vendor letter had been incorporated into maintenance procedure 12 MHP 5021.032.026 in a timely fashion.

ISSUE NO. 3: Drawing Revision Backlog

(Ref. Inspection Report Section 2.3.5, pg. 15)

Inspection Report Text (pg. 15)

"The licensee stated that the backlog of pre-1990 drawing changes...would be eliminated by January 1991 and a 60-day turnaround on all drawing revisions would be achieved by January 1992."

Clarification

We wish to clarify that the dates listed in the inspection report are internal AEPSC goals. A concentrated effort to meet these internal goals is being made. However, the goals should not be construed as NRC commitments.

ISSUE NO. 4: Inadequate Terminal Voltage at Steam-Driven AFW Pump Feedwater Inlet Valve

(Ref. Inspection Report Section 2.1.2, pg. 4 and Unresolved Item 90-201-04, pg. A-4)

Inspection Report Text (pg. A-4)

"The voltage drop calculation for the dc power feed cable to the steam-driven AFW pump feedwater inlet valve motor indicated that the worst-case terminal voltage at the motor was 178 Vdc. This condition could occur during a loss of ac power, with the station batteries at their end-of-life condition. The team could not determine if the 178 Vdc was sufficient for the valve to perform its required design function which is to control AFW flow. The team verified that vendor specification sheets of the valve only provided a single value of terminal voltage equal to 250 Vdc. Under these conditions the operability of the subject valve could not be verified."

Clarification

The NRC inspection team reviewed the circuit parameters for the AFW pump feedwater inlet valve motor. It was determined that the motor voltage terminal was 178 Vdc for the worst-case condition and the team questioned if this voltage was adequate. We had originally established the acceptability of the circuit by following a procedure described in the manufacturer's maintenance update letter. This procedure required that the inrush current be calculated and that it be a multiple of the full load current. For this type of circuit the motor's terminal voltage and the inrush current are proportional and when the inrush current was found acceptable it was expected that the terminal voltage would also be acceptable. However this procedure did not discuss a minimum acceptable voltage, and thus it could not be used to show to the NRC inspection team's satisfaction that the motor voltage was acceptable.

Following the inspection, an evaluation of the 178 Vdc was made by a different method. This method calculated the motor torque output at this reduced voltage against the required torque to operate the valve. For these types of motors the output torque is proportional to the motor terminal voltage and was derated by the ratio of rated voltage to actual voltage. Other factors in this method that were also considered were the valve operator gear efficiency and the gear ratio. The calculation for the involved valve showed that the motor produced an opening torque of 29.2 ft-lbs as compared to only 7 ft-lbs required to operate the valve. The motor voltage of 178 Vdc is therefore sufficient for this application.

ATTACHMENT 4 TO AEP:NRC:1138

COPY OF OVERHEADS FROM OCTOBER 4, 1990 MEETING

SSFI FOLLOW-UP MEETING
OCTOBER 4, 1990

I.	BACKGROUND	M. S. ACKERMAN
II.	DESIGN VERIFICATION	P. G. SCHOEPF
III.	TECHNICAL ISSUES	P. G. SCHOEPF J. R. ANDERSON
IV.	CONCLUSION	T. O. ARGENTA

ESW SSFI

REPORT RECEIVED AUGUST 24, 1990

- ESW ADEQUATE TO PERFORM ITS SAFETY FUNCTION
- CONCERNS WITH DESIGN VERIFICATION PROGRAM

LINKED WITH SEVERAL TECHNICAL ISSUES

HANDLING OF REPORT

REPORT DIVIDED INTO 53 SUBSECTIONS

- SPECIFIC AND GENERIC ITEMS

SUBSECTIONS ASSIGNED TO RESPONSIBLE GROUPS

- COMPUTERIZED TRACKING SYSTEM
- INDEPENDENT REVIEW OF RESPONSES

RESPONSES TO INCLUDE:

- INVESTIGATION
- CORRECTIVE ACTIONS
- PREVENTIVE ACTIONS

BACKGROUND (Cont'd)

SEPTEMBER 12, 1990

- NRC STAFF VISITED AEP OFFICES
- AEP INVITED TO DISCUSS REPORT WITH NRC MANAGEMENT

TOPICS

DESIGN VERIFICATION ISSUES

- SSFI FINDINGS
- HISTORICAL
- CORRECTIVE ACTIONS

SPECIFIC TECHNICAL ISSUES

- DESIGN PHILOSOPHY
- DESIGN ADEQUACY
- DECOUPLE TECHNICAL ISSUES FROM DESIGN VERIFICATION ISSUES

DESIGN VERIFICATION

- SUMMARY OF FINDINGS
- HISTORICAL PERSPECTIVE
- CONCLUSIONS

DESIGN VERIFICATION FINDINGS

- FAILURE TO ADEQUATELY IMPLEMENT NEW PROCEDURAL REQUIREMENTS
- INADEQUATE DESIGN VERIFICATION FOR CALCULATIONS
- METHOD OF INDEPENDENT DESIGN VERIFICATION FOR ORIGINAL PLANT DESIGN ACTIVITIES

DESIGN VERIFICATION FINDINGS

"CALCULATIONS PERFORMED AFTER THE NEW PROGRAM WAS IMPLEMENTED EITHER DID NOT RECEIVE AN INDEPENDENT DESIGN VERIFICATION OR RECEIVED AN IMPROPERLY COMPLETED ONE .... MANY POST-1988 CALCULATIONS WERE VERIFIED BY THE VERIFIER STATING NOT APPLICABLE ON THE INDEPENDENT VERIFICATION FORMS"

DESIGN VERIFICATION FINDINGS

EXAMPLE CALCULATIONS THAT DID NOT RECEIVE AN INDEPENDENT DESIGN VERIFICATION BUT WERE SUBSEQUENTLY VERIFIED:

- ESW FLOW REQUIREMENTS DURING A FIRE
- ESW PUMP ROOM TEMPERATURE
- ESW PUMP RUNOUT

COOK PLANT
MED RECORD - MED COPY
SECTION: PH&F
ENGINEER: MPS
DATE: 4/4/85
☒ PLANT LIFETIME
DATE TO PL...: N/A
☐ NON PERMANENT
MINIMUM RETENTION ___ YRS.

Sheet 1 of 22

△1 TOTAL 31 PAGES INCLUDING ALL ATTACHMENTS

AMERICAN ELECTRIC POWER SERVICE CORPORATION
MECHANICAL ENGINEERING DIVISION
PIPING, HVAC, & FIRE PROTECTION SECTION

CALCULATION CONTROL SHEET

CALCULATION NO. DCC HV12ES02N REVISION 0 SAFETY RELATED Y ✓ N ___

PLANT COOK UNIT NO. 2 SYSTEM ESW PUMP ROOM VENTILATION

TITLE HEAT GAIN / PRESSURE DROP CALC. — ESW PUMP ROOMS.

PURPOSE TO VERIFY THE ADEQUACY OF EXISTING VENTILATION SYSTEM FOR ESW PUMP ROOM BY DETERMINING IF THE MAX. OPERATING ROOM TEMP. IS WITHIN THE DESIGN CONDITION.

Revision	Author/ Date	Reviewer/ Date	MED Proc. 8 Check Method	Approval/ Date
0	M. R. Sanghani 3/10/89	H.W. Young 3/13/89	1	[signature] 3/14/89
1	M. R. Sanghani 6/20/90	H. W. Young III 6/21/90	SEE VERIFICATION CHECKLIST ATTACH. - XI.	[signature] 6/26/90

CALCULATION CHECK METHODS PER MED 8, PARAGRAPH 4.3

1. Check of Input Data, Method & Numerical Accuracy
2. Independent calculation
3. Comparison with the results of a similar calculation
4. Engineering judgement that Input, Method & Results are reasonable
5. Test Case to verify Computer Program

Notes: ① THIS CALC. SUPERSEDES CALC. # DCCHV12ES-01-N.
② S.S.F.I. - ITEM # SMK-9.
③ REVISION 1 ITEMS ARE IDENTIFIED WITH △1

Microfilm Required Y ✓ N ___ Microfilm No. ___

Cross Reference File ___

Attachment 2

NED CALCULATION VERIFICATION CHECKLIST

Reviewer: P.O. Chiusano Date: 6/15/90

Section Manager Approval: [signature] Date: 9/14/90

HXP900613AF

The review of the calculation shall include evaluation against the following questions:

	Yes	No	Basis For Determination
1. Was an appropriate method used?	✓	___	
2. Are the results reasonable compared to the input?	✓	___	
3. Are the results numerically correct?	✓	___	
4. Are the equations used correct and the reference documented?	✓	___	Crane Tech. Paper No. 410 Cameron Hydraulic Data
5. Were the correct inputs used and their sources documented?	✓	___	
6. Are the assumption(s) reasonable?	✓	___	
7. Does the calculation include the necessary documentation as described in NEP 6.4, Sections 5.0 and 6.0?	✓	___	NEP 6.4
8. Is the calculation deemed acceptable?	✓	___	

NEF-6.4A
(5/89)

CALCULATION ANALYSIS

Nuclear Engineering Department

CC-1-3.18

HEP & T SECTION SHEET 1 OF 6

I.D. NO. HXP900613AF	PLANT Cook Nuclear UNIT 1 & 2
SAFETY RELATED YES ✓ NO ___ SYSTEM ESW	COMPANY I & M
TITLE ESW PC Runout	CALCULATED BY: [signature] 6/13/90 DATE
FILE LOCATION 24.1.5	CHECKED BY: R.D. Chiasson 6/15/90 DATE
MICROFILM NO. ___	APPROVED BY: [signature] 6/15/90 DATE

PROBLEM DESCRIPTION: Determine if an ESW pump can runout due to a break occurring on the return piping at the exit to the turbine building.

DESIGN BASIS OR REFERENCES: References noted in calc.

METHOD OF VERIFICATION: NEP 6.4 section 7.3.1 Alternate calculation, Attachment 1, 5 pages. Attach 2, 1 page

REVISIONS

NO.	REASON FOR CHANGE	PREP'D BY	DATE	CKD. BY	DATE	APVD. BY	DATE

COOK NUCLEAR PLANT
NED RECORD-NED COPY
SECTION HEP&T
ENGINEER [signature]
DATE 6-22-90
☒ PLANT LIFETIME
DATE TO PLANT N/A
☐ NON PERMANENT
MINIMUM RETENTION ___ YRS.

METHOD OF VERIFICATION: ___

NEF64A.DOC
C:\FORM,D-E,FF-363

7.4.1 Discrepancies and deviations shall be resolved by the reviewer and preparer. A copy of these resolved comments should be kept.

7.4.2 If the reviewer and preparer cannot resolve the discrepancy, the calculation shall be submitted to the Cognizant Section Manager for resolution.

7.5 After the review is complete, the reviewer shall sign and date the cover sheet of the calculation package and submit the calculation to the Cognizant Section Manager or designated alternate for approval. Each page of the calculation may show reviewer's initials if the format of the page has space for this information.

7.6 The Cognizant Section Manager shall approve the calculation and the Verification Checklist, and as a minimum shall sign and date the cover sheet, signifying approval, and return the calculation to the preparer. Each page of the calculation may be approved if the format of the page has space for this approval. CS-1

7.7 Copies of the approved calculation package shall be distributed to any AEPSC Section and/or Plant Department with a "need to know" (Further studies required, equipment application/operation impact, regulatory guide compliance, etc.) as identified in paragraph 6.9.

7.8 A copy of the calculation shall be placed in the file that is provided for that type of calculation.

7.9 The calculation shall be processed for lifetime record retention in accordance with NEP 17.0.

8.0 REVISIONS

8.1 Calculation revisions shall be prepared, reviewed, approved, distributed and controlled in the same manner as original calculations.

9.0 REFERENCES

GP 18.0
GP 18.1
NEP 2.6
NEP 3.4
NEP 3.5
NEP 17.0
NEP 17.2

DESIGN VERIFICATION FINDINGS

"...INDEPENDENT DESIGN VERIFICATIONS WERE EITHER NOT PERFORMED OR WERE PERFORMED INADEQUATELY FOR ORIGINAL PLANT DESIGN CALCULATIONS, TEST REPORTS AND DRAWINGS, AND DESIGN CHANGES AND DESIGN ACTIVITIES SUBSEQUENT TO PLANT LICENSING"

NRC DESIGN CONTROL INSPECTION REPORT

- PROCEDURES ADEQUATE TO ENSURE VERIFICATION MET REQUIREMENTS OF ANSI N45.2.11
- "DESIGN VERIFICATION ACTIVITIES CONDUCTED DURING ORIGINAL PLANT DESIGN WERE CONSISTENT WITH GENERALLY ACCEPTED ENGINEERING PRACTICE FOR THE TIME THEY WERE ACCOMPLISHED"

REPORT NOS. 50-315/84-16, 50-316/84-18

DESIGN CONTROL
AEPSC REORGANIZATIONS

- MERGED ELECTRICAL AND MECHANICAL ENGINEERING INTO A SINGLE NUCLEAR ENGINEERING DEPARTMENT (1988)
- CONSOLIDATED DESIGN SUPPORT FUNCTIONS INTO A NUCLEAR DESIGN GROUP (1989)
- DEVELOPING IMPLEMENTING PROCEDURES TO ACCOMPLISH "NUCLEAR" ACTIVITIES

DESIGN VERIFICATION CONCLUSIONS

- SOME ACTIVITIES CITED AS NOT HAVING VERIFICATIONS DID RECEIVE APPLICABLE REVIEWS
- FINDINGS WERE NOT SAFETY SIGNIFICANT
- REQUIREMENTS HAVE BEEN AUGMENTED
- ORIGINAL PRACTICES WERE PREVIOUSLY ACCEPTED

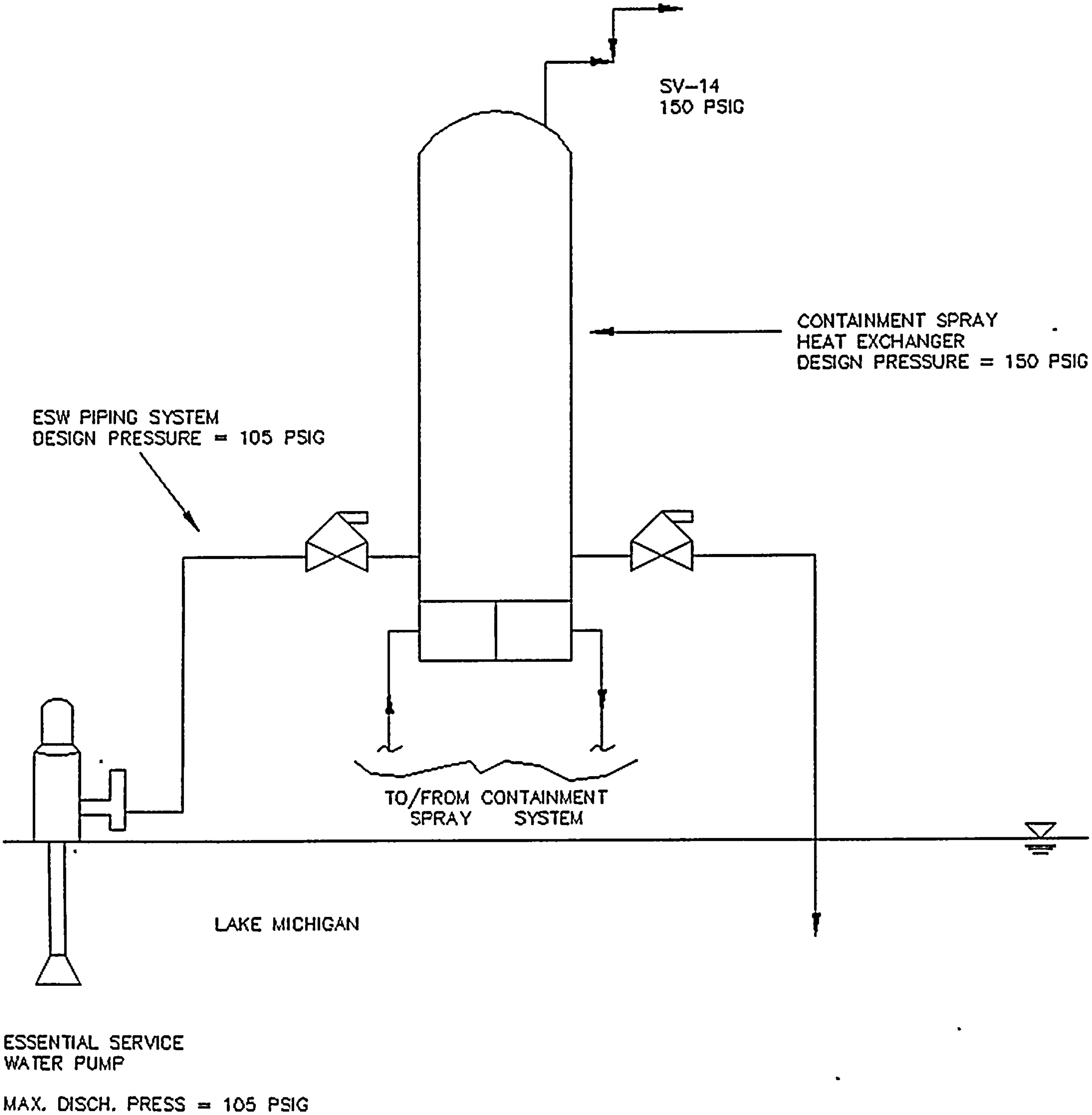
TECHNICAL ISSUES

- ESW THERMAL RELIEF VALVES
- CABLE SIZING
- BREAKER RATINGS
- INVERTER RATING
- DIESEL GENERATOR SIZING
- BATTERY TESTING

THERMAL RELIEF VALVES FINDING

"DESIGN DOCUMENTATION INDICATED THAT THE THERMAL RELIEF VALVES ON SEVERAL HEAT EXCHANGERS HAD BEEN SET AT 150 PSIG. THIS WAS INCONSISTENT WITH ASME B&PV CODE WHICH REQUIRED THAT THE SET PRESSURE OF THE RELIEF VALVE BE AT, OR LOWER THAN THE DESIGN PRESSURE. THE LICENSEE STATED THAT THE RELIEF VALVE SETPOINTS WOULD BE LOWERED TO 105 PSIG."

CONTAINMENT SPRAY HEAT EXCHANGER THERMAL RELIEF VALVE



ESW THERMAL RELIEF VALVES

- ESW PIPING SYSTEM DESIGN PRESSURE IS 105 PSIG (PUMP SHUTOFF HEAD)
- COMPONENTS SPECIFIED AT 150 PSIG, WHICH IS CONSERVATIVE
- THERMAL RELIEF VALVE SET PRESSURE SET AT 150 PSIG CONSISTENT WITH COMPONENT DESIGN PRESSURE
- WORDING CHANGES IN SECTION XI REQUIREMENTS POINTED TO HYDRO TESTS BASED ON RELIEF VALVE PRESSURE RATHER THAN DESIGN PRESSURE
- SELF-IDENTIFIED RESOLUTION TO LOWER VALVE SET PRESSURE TO SYSTEM PRESSURE FOR TESTING CONVENIENCE
- CHARACTERIZED AS EXAMPLE OF INADEQUATE DESIGN VERIFICATION CONTRIBUTING TO DESIGN DEFICIENCIES

ELECTRICAL ISSUES

- TECHNICAL DISAGREEMENTS NOT PROBLEMS
  - DESIGN BASIS ACHIEVED IN ALL CASES
  - DESIGNS CONTROLLED AND REVIEWED
  - CONSERVATISMS CHALLENGED
  - MARGINALLY RELEVANT ISSUES INCLUDED
- ELECTRICAL EXAMPLES
  - ELECTRICAL CABLE SIZING
  - 4KV CIRCUIT BREAKER RATING
  - INSTRUMENTATION INVERTER VOLTAGE QUALIFICATION
  - EMERGENCY DIESEL GENERATOR SIZING
  - STATION BATTERY TESTING

ELECTRICAL CABLE SIZING

- ISSUE - CABLE FAULT DUTY
  - CABLE INSULATION DAMAGE
- DESIGN BASIS ACHIEVED
  - CABLE SIZED PER IPCEA AMPACITY TABLES
- SIZING PROCEDURALLY CONTROLLED
  - AMPACITY
  - VOLTAGE DROP
- ECONOMIC NOT TECHNICAL ISSUE
  - 3-PHASE FAULT REQUIRED AT MID CABLE
  - FAULTED CABLES REPLACED, NOT REPAIRED
  - CABLES TESTED FOR END DEVICE FAULTS
- CONCLUSION
  - ISSUE IS NOT AN EXAMPLE OF INADEQUATE DESIGN VERIFICATION
  - NO FURTHER ANALYSIS REQUIRED

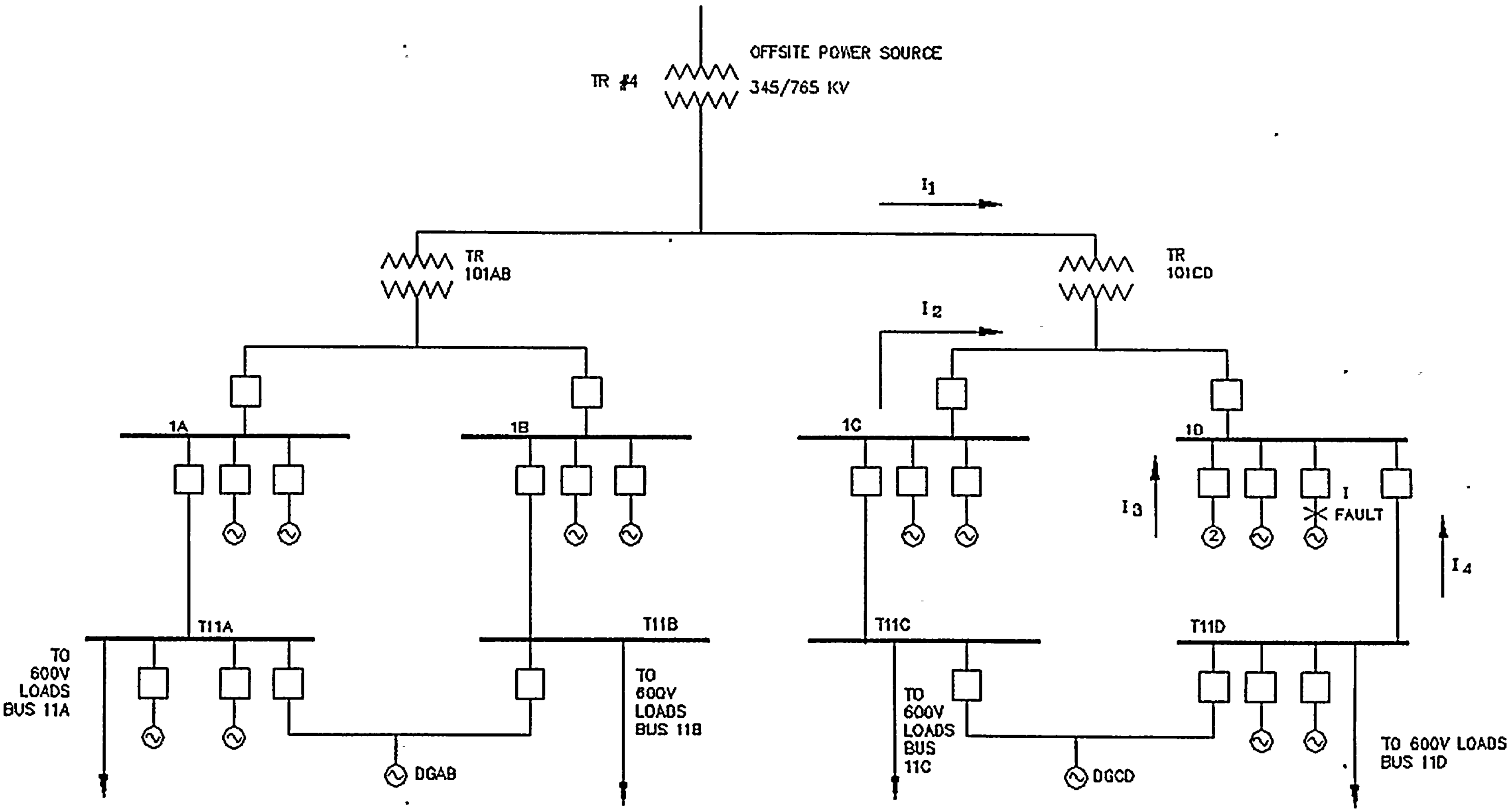
CIRCUIT BREAKER RATING

- ISSUE - 4KV CIRCUIT BREAKER UNDERSIZED

- DESIGN BASIS ACHIEVED
  - ORIGINAL PLANT DESIGN SIZING
  - IEEE 279 SINGLE FAILURE ANALYSIS
  - METHOD OF FAULT CURRENT COMPUTATION

- POTENTIAL DEFICIENCY RECOGNIZED - 1988
  - DESIGN BASIS REVIEWED

4KV BREAKER FAULT CURRENT



ASSUME FAULT ON BUS 1D BREAKER LOAD SIDE TERMINALS SINCE THIS IS THE WORST CASE BUS MOTOR CONTRIBUTION.

CIRCUIT BREAKER RATING cont'd

- ENGINEERING ANALYSIS - VERY LOW PROBABILITY EVENT
  - MAXIMUM DC OFFSET
  - BOLTED 3-PHASE FAULT
  - FAULT AT OR NEAR BREAKER
  - ATYPICAL SYSTEM CONFIGURATION
- ENGINEERING ANALYSIS - PLANT DESIGN FEATURES
  - METAL ENCLOSED SWITCHGEAR
  - SWITCHGEAR NOT ACCESSED WHILE ENERGIZED
  - INDIVIDUAL SHIELDED, GROUNDED, PHASE CABLES

CIRCUIT BREAKER RATING cont'd

- GROUNDED SYSTEM 2,000 AMPS MAXIMUM FAULT CURRENT
- SWITCHGEAR WITHIN INTERRUPT RATINGS

° CONCLUSION

- 1988 REVIEW SUPPORTS ORIGINAL DESIGN PHILOSOPHY
- NOT AN INADEQUATE DESIGN VERIFICATION PROBLEM
- ISSUE NOT A PROBLEM

EMERGENCY DIESEL GENERATOR SIZING

- ISSUE - UNCONSERVATIVE EDG SIZING CALCULATION
- DESIGN BASIS ACHIEVED
  - DYNAMIC STUDY COMPLETED - 1972
  - DESIGN BASIS REVIEW - 1988
- ENGINEERING REVIEW
  - CALCULATION CONTROLLED AND VERIFIED
  - CONSERVATIVE ASSUMPTIONS
    - ALL 4KV MOTORS AT MAXIMUM LOADING
    - MOTORS AT 90% EFFICIENCY
    - DIVERSIFICATION USED
      - HEAT TRACING
      - EDG ROOM HEATERS
      - EDG AIR COMPRESSORS
      - AUX BUILDING ELEVATOR

EMERGENCY DIESEL GENERATOR SIZING

- ENGINEERING CALCULATION DURING AUDIT
  - ELIMINATED DIVERSIFICATION
  - EDG WITHIN 1% OF CONTINUOUS RATING
  - EDG WELL WITHIN 2,000 HOUR RATING

- CONCLUSION
  - DOCUMENTED CONSERVATIVE ENGINEERING JUDGEMENT
  - CONSERVATIVE LOAD SERVICE FACTORS
  - EDG IS WITHIN RATINGS

INSTRUMENTATION INVERTER VOLTAGE QUALIFICATION

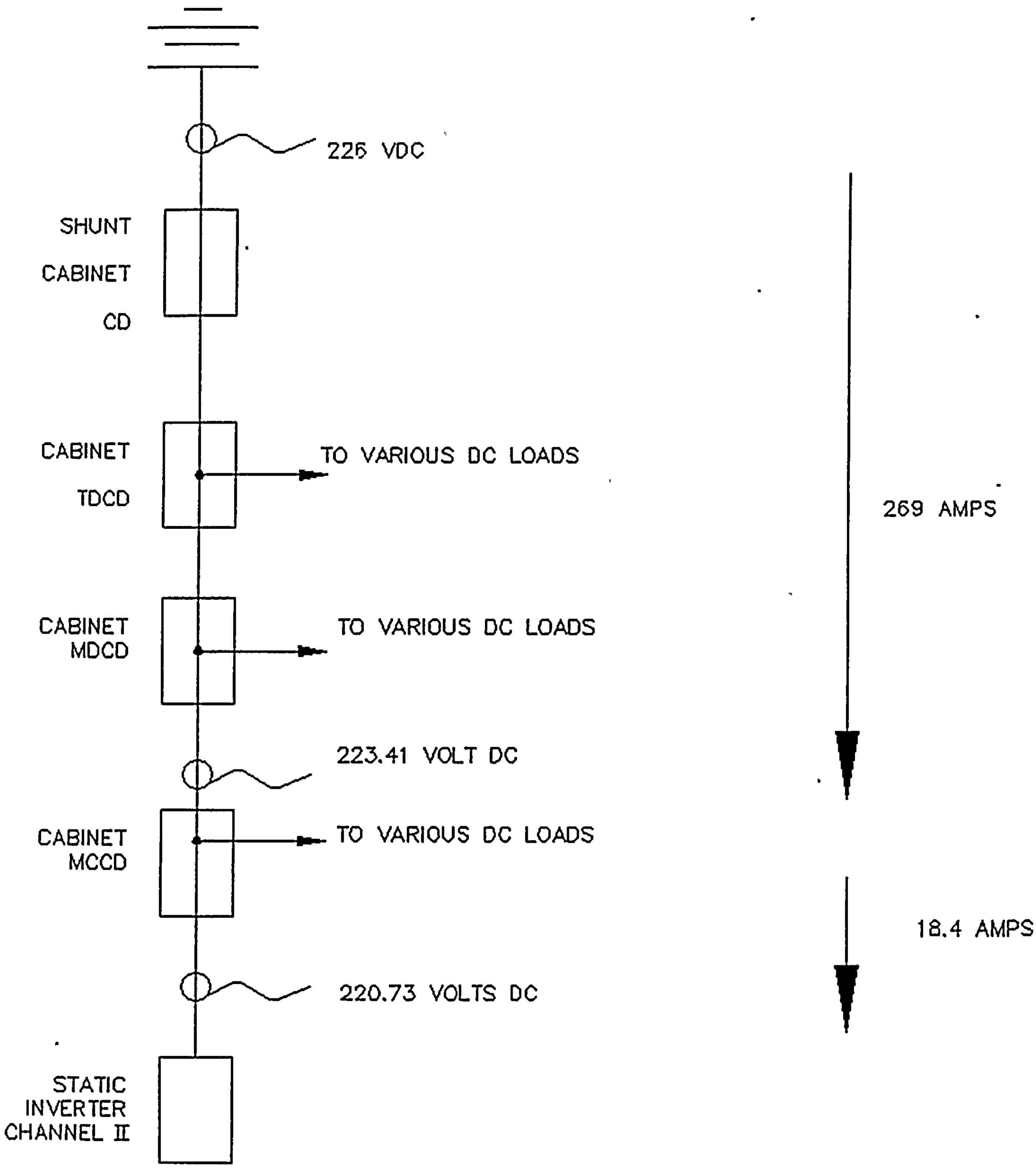
- ISSUE - END OF LIFE BATTERY VOLTAGE SAME AS MINIMUM INVERTER VOLTAGE RATING 210 V D.C.

- DESIGN BASIS ACHIEVED
  - INVERTERS REQUIRED - 4 HOURS
  - BATTERY VOLTAGE - 226 V, INVERTER VOLTAGE - 220V

- ENGINEERING REVIEW
  - BATTERY vs INVERTER DUTY
  - VOLTAGE DROP REDUCED AT END OF PROFILE
  - INVERTER MANUFACTURER - 200 VOLT FUNCTIONAL STATEMENT

- CONCLUSION
  - INVERTER APPLIED WITHIN RATINGS
  - NOT A DESIGN VERIFICATION PROBLEM

STATIC INVERTER VOLTAGE PROFILE



STATION BATTERY TESTING

- o ISSUE - BATTERY TESTING INADEQUATE

- o DESIGN BASIS ACHIEVED
  - CAPACITY TEST
  - PERFORMANCE TEST - CONNECTED LOADS

- o BATTERY TESTING IMPROVEMENTS
  - IEEE 450
  - COMPUTERIZED LOAD BOX
    TECHNICAL SPECIFICATION REVISION
  - 3 OF 4 STATION BATTERY'S NOW TESTED

- o CONCLUSION
  - TECH SPEC REQUIRED TESTING
    <u>WAS</u> PERFORMED
  - TESTING IMPROVEMENTS IN PROGRESS

CONCLUSIONS

- SSFI REPORT RAISED VALID POINTS
  - CORRECTIVE ACTION TAKEN FOR DESIGN VERIFICATION
- DESIGN VERIFICATION ISSUES SEPARATE FROM TECHNICAL ISSUES

ACTIONS FOLLOWING SSFI DESIGN VERIFICATION FINDINGS

- REVIEWED EXAMPLE SSFI FINDINGS WITH ENGINEERS IN FORMAL TRAINING SESSION
- DIRECTED HISTORICAL REVIEW OF DESIGN OUTPUTS PERFORMED TO NEW PROCEDURES
- FORMED QUALITY REVIEW TEAM TO REVIEW FUTURE DESIGN OUTPUTS
- PLANNED SUBJECT TRAINING IN ALL AREAS OF DESIGN CONTROL
- SENIOR MANAGEMENT DIRECTIVE ON PROCEDURAL ADHERENCE

FUTURE PLANS

YEAR	SYSTEM
1991	(1) ELECTRICAL DISTRIBUTION & DIESEL GENERATORS
	(2) VENTILATION: CONTROL ROOM, ENGINEERED SAFEGUARDS, SPENT FUEL POOL
1992	CONTAINMENT SPRAY
1993	SSOMI & SDFI FOLLOWUP
1994	COMPONENT COOLING WATER
1995	EMERGENCY CORE COOLING