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AEP:NRC:0773AB

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
REGULATORY GUIDE 1.97, REVISION 3

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

Attn: T. E. Murley

October 5, 1988

Dear Dr. Murley:

The purpose of this letter is to provide you with an updated status report (attached) that details our compliance with Revision 3 of Regulatory Guide 1.97. In our previous submittal (AEP:NRC:0773AC, dated June 20, 1988), we had committed to providing you this updated status report by October 1, 1988. This status report consolidates, clarifies and updates all of our previous submittals (AEP:NRC:0773O, dated October 15, 1985 and AEP:NRC:0773S, dated June 29, 1987). We again request that your final licensing decision concerning Regulatory Guide 1.97, Revision 3 be based on the attached updated status report.

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

Sincerely,

M. P. Alexich
Vice President

MPA/eh
Attachment
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AEP:NRC:0773AB
DC-N-6015.1

ATTACHMENT TO AEP:NRC:0773AB

STATUS REPORT

IMPLEMENTATION PLAN OF REGULATORY GUIDE 1.97, REVISION 3,

FOR THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2

1.0 BACKGROUND

In August 1984, American Electric Power Service Corporation (AEPSC) contracted the engineering consulting firm of DiBenedetto, Farwell & Hendricks to perform the detailed design study required to determine the status of the Cook Nuclear Plant Units 1 and 2 compliance with Regulatory Guide 1.97, Revision 3. A preliminary Status Report (AEP:NRC:0773J dated February 28, 1985), based on preliminary findings by our consultant, was submitted to the NRC staff. A Final Status Report (AEP:NRC:0773O dated October 15, 1985) was then issued based on further work done by our consultant, with the understanding that further updating may be necessary. We subsequently issued an update to our October 15, 1985, submittal (AEP:NRC:0773S dated June 29, 1987) that responded to numerous questions asked by the NRC in a Preliminary Technical Evaluation Report and also submitted various specific deviations from Regulatory Guide 1.97, Rev. 3 recommendations. In order to consolidate, clarify, and update our previous submittals, AEP:NRC:0773J, AEP:NRC:0773O, AEP:NRC:0773S, we are submitting this document for NRC review and final licensing action.

It should be noted that because design change activities have been ongoing since 1986 with respect to Regulatory Guide 1.97, some planned actions identified in AEP:NRC:0773O have since been completed. These items are identified within Section 3.0 of this letter as "No Further Action Required" in the Remarks/Action Req'd column and listed as "CMPLT" in the Unit 1 or Unit 2 Schedule column.

2.0 STATUS REPORT

Section 3.0 of this attachment contains information regarding instrument range, environmental qualification, seismic qualification, quality assurance, redundancy, power supply, location of display, remarks, and, as appropriate, an upgrade schedule for each type A,B,C,D and E variable listed in Regulatory Guide 1.97, Revision 3. The format and content of Section 3.0 is

consistent with the requirements of Section 6.2 of Supplement No. 1 to NUREG-0737. Section 3.0 is also consistent in organization with Table 3 (PWR Variables) of Regulatory Guide 1.97, Rev. 3 dated May 1983. The schedule for each instrument indicates, as applicable, when the recommendations of Regulatory Guide 1.97, as described in this attachment, will be met.

In those instances in which the design of our post-accident monitoring (PAM) systems deviate from the guidance provided in Regulatory Guide 1.97, Revision 3, these deviations are explicitly identified in the Section 3.0 tables, consistent with the requirements of Section 6.2 of Supplement 1 to NUREG-0737. A discussion of each deviation is contained in Section 2.1 and 2.2. A summary of deviations is provided in Section 2.3. The Section 2.3 summary also cross-references the deviations to Section 3.0 table entries.

2.1 DEVIATIONS RELATED TO LICENSED DESIGN

This section provides a discussion of those areas in which we have identified deviations from the Regulatory Guide 1.97, Revision 3 recommendations where the deviations are primarily a result of the originally licensed design of the Cook Nuclear Plant. A deviation number is assigned to each deviation identified; these deviation numbers are used in the Section 2.3 summary and the Section 3.0 table entries.

2.1.1 Deviation No. DV-1; Environmental Qualification

As provided by 10 CFR 50.49(k), originally installed Cook Nuclear Plant Qualified instrumentation located in potentially harsh environment has been qualified in accordance with "Guidelines for Evaluating Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines). Qualified equipment ordered after February 22, 1983, is to have environmental qualification in accordance with Category I of NUREG-0588 (i.e., IEEE Std. 323-1974) unless there are sound reasons to the contrary.

Equipment located in a mild environment is not required to be environmentally qualified. The above is consistent with the current licensing basis of the Cook Nuclear Plant. However, because we do not literally comply with the Reg. Guide 1.97 recommendations for environmental qualification, we are noting herein our deviation from the Reg. Guide 1.97 guidance on this issue.

2.1.2 Deviation No. DV-2; Seismic Qualification

Seismically qualified equipment meets the provisions of Cook Nuclear Plant Updated FSAR p. 7.2-12 dated July 1982 (i.e., protection equipment is designed such that, for a design basis earthquake, the equipment will not lose its capability to perform its design objective, to shut the plant down and/or maintain the unit in a safe shutdown condition). Reactor protection Instrumentation originally installed at the Cook Nuclear Plant was seismically tested by Westinghouse Electric Corporation as documented in WCAP-7397-L, "Topical Report Seismic Testing of Electrical and Control Equipment" dated January 1970. No industry standards regarding seismic qualification existed at that time. Consistent with our current licensed design, equipment ordered for Regulatory Guide 1.97 upgrading will be seismically qualified in accordance with IEEE Std. 344-1975 unless there are sound reasons to the contrary. Seismic qualification for existing Category 1 circuits is provided from the sensor up to and including the channel isolation device (shown as a "Signal Isolator" in Figure 2.1-1), typically installed in the reactor protection cabinets located in the control room area. Our design does not provide for seismic qualification of equipment beyond the channel isolation device for existing Category 1 circuits; however, we have installed seismically qualified indicators and/or recorders on variables monitored by Category 1 instrumentation. This was done because we believe that the primary indicators/recorders that provide direct Category 1 variable indication are the only credibly vulnerable equipment installed beyond the isolators that could provide ambiguous or misleading information due to a failure during a seismic event.

It should be noted that other equipment beyond the channel isolation device (such as control equipment) that directly provides information for the channel indication is of a similar design to existing equipment that is seismically qualified (such as reactor protection instrumentation). As such, in spite of the fact that we do not take credit for seismic qualification, we believe that this equipment would not provide ambiguous or misleading information to the operator following a seismic event.

Our design does not provide seismic qualification of cables and equipment within the non-safety related portion of the PAM circuits. Nevertheless non-safety related portions of the PAM channels that directly provide PAM indication are located in the control room, or control room vault, which are seismically qualified structures. Cables and equipment associated with PAM circuits located outside cable vault areas (e.g., cables to valves, cables to Technical Support Center (TSC) I/O cabinets) may not be maintained entirely in a seismically qualified structure.

Since our current licensing basis, as described above, does not literally comply with the Reg. Guide 1.97 recommendations for seismic qualification beyond the isolation device for existing circuits, we are noting herein our deviation on this issue. Figure 2.1-1 provides an example of an existing PAM channel that is typical of our originally licensed plant design.

Consistent with the Cook Nuclear Plant licensing basis, PAM instrument channels or portions thereof scheduled to be upgraded or added per Reg. Guide 1.97 and requiring seismic qualification will be installed to meet Reg. Guide 1.97 Category 1 recommendations except as noted above and in the tables in Section III.

2.1.3 Deviation No. DV-3; Quality Assurance

The provisions of the Cook Nuclear Plant QA program as described in Updated FSAR Section 1.7 have been applied to the safety-related

portions of the PAM circuitry. This program satisfies the requirements of 10 CFR 50, Appendix B. The implementation of specific Regulatory Guides and ANSI Standards regarding quality assurance is consistent with the commitments of Cook Nuclear Plant FSAR Section 1.7, Appendix A, dated July 1988, which address all but one (Regulatory Guide 1.28) of the guidance documents referenced by Reg. Guide 1.97, Rev. 3. The Cook Nuclear Plant QA program has not necessarily been applied to non-safety related portions of the PAM system (see Fig. 2.1-1).

Since we do not literally comply with the Reg. Guide 1.97, Rev. 3 quality assurance recommendations, we have noted herein our deviations from the Reg. Guide guidance in this area.

2.1.4 Deviation No. DV-4; Redundancy

Instruments installed at the Cook Nuclear Plant to meet redundancy requirements have a minimum of two (2) redundant, electrically independent and physically separate channels up to and including any isolation device (shown as "Signal Isolator" in Figure 2.1-1) typically installed in the reactor protection cabinets located in the control room area. For existing circuits, the isolation device is providing isolation between safety-related circuits and non-safety related circuits, which includes portions of the PAM circuitry as per our licensed design. The display may be a common multi-pen recorder or a dual indicator. Separation of safety-related circuits up to and including the isolation device is in accordance with Updated FSAR p. 7.2-4 dated July 1982 .

Figure 2.1-1 is an electrical schematic that shows a typical cable and hardware configuration for redundant PAM channels. This illustration presents a typical configuration and is not meant to represent an actual circuit. Exact circuit configuration may vary from the illustration. This configuration reflects our originally licensed design. This design was completed prior to the issuance of the NRC guidance regarding physical independence of electrical

systems (Regulatory Guide 1.75, September 1978) and the associated IEEE standard for separation of Class 1E equipment and circuits (IEEE Std. 384-1974).

Physical separation (of both electrical cabling and hardware) is maintained between the redundant reactor protection set channels. The BOP interconnecting cabling between the reactor protection cabinets and the various PAM readout devices is routed through the cable vault area and then back to the control room. The redundant PAM signal and power cables in the cable vault area and in the control room are not physically separated. Also, the redundant PAM readout devices are not physically separated. As previously stated, there are also some cases of redundant PAM signal cabling feeding a single readout device (recorder or computer).

At the request of NRC staff, an evaluation of the impact of this lack of physical separation in these areas was performed. This evaluation identified two events that have the potential for compromising the integrity of the PAM system as presently installed. These events are a fire in the cable vault area or a severe natural phenomenon. In the case of a severe natural phenomenon, the only event of significance with regard to the cable would be an earthquake. Even in the unlikely event of a design-basis earthquake, the PAM system will continue to serve its intended function, after the appropriate system modifications have been completed. In the event of a fire in the cable vault area, we would not expect to need the PAM instruments to follow the course of another accident, and the ability to bring the plant to a safety configuration would not be compromised. This position is consistent with actions we have taken to ensure compliance with Appendix R to 10 CFR 50, "Fire Protection Program for Nuclear Power Facilities Operating prior to January 1, 1979."

Figure 2.1-1 also shows cables leaving the control room and vault areas and going out into other plant areas. These areas in most cases will not offer the same missile and fire protection as offered

by the control room and vault areas. The signals going to these remote areas are also not isolated from the PAM signals in the control room or cable vault areas. The cables in most cases have been installed in tray and conduit which has been mounted to the same requirements as safety-related tray and conduit. The cable is the same as that used for safety-related circuits. The redundant PAM signal cables are not physically separated. In some cases, cables carrying redundant signals also terminate in a cabinet or device that has not been designed to the same separation requirements as safety-related devices. These cabinets or devices may not be seismically qualified.

Consistent with the current licensing basis of the Cook Nuclear Plant PAM instrument channels or portions thereof scheduled to be upgraded or added per Reg. Guide 1.97 and requiring redundancy will be installed to meet Category 1 recommendations except as noted above and in the tables in Section III.

Since our current license design as described above does not conform to the Reg. Guide 1.97, Rev. 3 recommendations for physical separation beyond the isolation devices installed in PAM circuits, we are noting herein our deviations from the Reg. Guide guidance on this issue.

2.1.5 <u>Deviation No. DV-5; Display Location</u>

The display will be indicated either inside the control room or at another location as permitted by Regulatory Guide 1.97. In some instances, the analog recording of Category 1 variables is not directly provided, however the Technical Support Center computer does record these variables when analog recording is not provided. Because we do not literally comply with the recording recommendations of Reg. Guide 1.97 Rev. 3 for Category 1 variables, we are noting herein our deviation from the Reg. Guide guidance on this issue. Please note that this applies to analog variables only. Discrete variables such as breaker or valve positions are not recorded.

2.2 DEVIATIONS IDENTIFIED FOR SPECIFIC VARIABLES

2.2.1 Deviation No. DV-6; Containment Pressure

Monitoring of containment pressure is currently provided by two Category 3 wide-range (-5 to 36 psig) instruments and four Category 3 narrow-range (-5 to 12 psig) instruments. The design pressure of the Cook Nuclear Plant containments is 12 psig. The four narrow-range instruments are scheduled to be upgraded to meet Reg. Guide 1.97 Category 1 recommendations by the end of the 1987 refueling outage for Unit 1 (which has been completed) and by the end of the 1988 refueling outage for Unit 2.

The wide-range containment pressure instrumentation ranges were revised to meet the requirements of NUREG-0578 and NUREG-0737. However, these instruments are not powered by an emergency standby power source as recommended by Regulatory Guide 1.97, Rev. 3 for Category 1 instrumentation, and they do not meet the Category 1 separation criteria. The wide-range instrumentation is, however, highly reliable, and as a result we believe it is unlikely that it would not be available if needed to monitor the course of an accident. Further, for other than short-term individual compartment pressure peaks, the narrow-range instrumentation would span the range of pressure anticipated in our evaluation of loss-of-coolant type accidents. The above justification is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 wide-range containment pressure instrumentation.

2.2.2 Deviation No. DV-7; Subcooling Meter

The saturation meter equipment was originally installed in accordance with the requirements of NUREG-0578. In an SER dated March 20, 1980, the equipment installed to monitor degrees of subcooling was found to be acceptable (NRC letter, A. Schwencer to John E. Dolan, dated March 20, 1980). As noted in that correspondence, the device installed was a discrete digital monitor,

and the plant process computer was used in conjunction with this monitor to provide subcooling margin. Additionally, as part of the NUREG-0737 Supplement 1 requirements, a subcooling margin is provided by our Technical Support Center computer. We believe that these three instrument systems, which have been installed to be consistent with the requirements of their appropriate documents, are reliable. As a result, we believe it is unlikely that they would not be available if needed to monitor the course of an accident. Further, neither NUREG-0578, nor subsequently, NUREG-0737 required seismic qualification or redundancy for this instrumentation.

The above information is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 instrumentation for monitoring degrees of subcooling. (This deviation is also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.3.)

2.2.3 Deviation No. DV-8; Reactor Coolant System (RCS) Sampling/Boron Concentration Range

Primary coolant boron concentration can be measured in the range of 375 ppm to 10,000 ppm. This range is based on PASS reactor coolant samples with a 1:1000 dilution.

The PASS would be used during and following loss-of-coolant accidents. In the event of a LOCA, emergency boration and injection from the refueling water storage tank would occur and we would therefore expect a reactor coolant boron concentration substantially in excess of the low range of our PASS sample measurement capability.

The above is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide the capability to measure boron concentration in PASS samples to the lower limit of 0 ppm. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.2.)

2.2.4 Deviation No. DV-9; RCS Sampling/Radioactivity

As stated in our original submittal (AEP:NRC:0773O, dated October 15, 1985), the primary coolant system radioactivity is not continuously monitored by in-line instrumentation. Rather, periodic analysis of reactor coolant grab samples is provided to detect deterioration of fuel cladding. Our post-accident sampling system provides a diluted grab sample that is analyzed by the gamma spectrum analyzer in a range of 1 uCi/ml to 10 Ci/ml. This measurement range is consistent with the Reg. Guide 1.97, Rev. 3 recommendations for this parameter. The above information is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for continuous monitoring of radioactivity in the reactor coolant system. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.5.) It should also be noted that Category 1 requirements for this system are only to be applicable to equipment that operates equipment installed in the portion of piping that is Seismic Class I. Electrical equipment operating equipment installed in Seismic Class 3 piping is to meet Category 3 requirements.

2.2.5 Deviation No. DV-10; Quench Tank Level

We do not rely on the quench tank to perform any post-pressurizer release function. However, we are providing the following information in response to the evaluation of our October 15, 1985, submittal (AEP:NRC:0773O) performed by EG&G on behalf of the NRC.

The range of 74% of total tank volume originally submitted was not accurately stated to show the adequacy of the existing installation. The correct range should have been stated as being from 7 inches above the tank bottom to 7 inches below the tank top. This range includes coverage of the sparger. With regard to the ability to quench a "design-basis" pressurizer release, as noted above we do

not rely on the quench tank to perform this function. The quench tank is used during normal plant operation to contain pressurizer releases from routine pressurizer pressure adjustments and valve leakage. In the case of a design-basis event that causes the PORVs and safety relief valves to lift, two rupture discs will burst before reaching the quench tank design pressure of 100 psi. Subsequently, discharge through the quench tank into the containment sump will occur.

With regard to over pressurization, we do not understand the basis for the EG&G position that sufficient gas volume exists to accept pressurizer release without becoming over pressurized. As noted above, over pressurization will not occur because rupture discs will burst and discharge into the containment before reaching the tank design pressure of 100 psig.

Normal water level is kept at between 80% and 84% of the instrument range with a high alarm at 84% and a low alarm at 79%. As such, in-leakage from the relief discharge system can be adequately monitored. The above information is the basis for our deviation from the Regulatory Guide 1.97 recommendation to monitor quench tank level from top to bottom of the tank. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.15.)

2.2.6 Deviation No. DV-11; S/G Wide-Range Level

On August 21, 1981, we submitted a letter (AEP:NRC:0300G) that documented discussions with NRR staff clarifying certain portions of an NRC SER (June 16, 1981) of the Cook Nuclear Plant auxiliary feedwater system. In that letter it was confirmed that Regulatory Guide 1.97 recommendations for steam generator level instrumentation did not have to be implemented at that time, but that implementation would be addressed at some time in the future through the Regulatory Guide 1.97 compliance/commitment process.

The steam generator wide-range level indication is not required for post-accident monitoring and in fact has been deleted from our

Technical Specifications for Units 1 and 2. This was stated in our December 10, 1980, letter, which submitted a proposed amendment to our Technical Specifications (AEP:NRC:0449). As stated in that letter, the reasons for deletion of steam generator wide-range level indication from the Technical Specifications are: (1) the S/G wide-range level indication does not perform any safety-related function and is not assumed operable in the various plant safety analyses; and (2) the S/G narrow-range instrumentation, which we believe fulfills post-accident monitoring requirements, is environmentally and seismically qualified, powered from a Class 1E source and has three redundant channels per S/G. The S/G level indication is backed up by auxiliary feedwater flow instrumentation.

The S/G wide-range level instrumentation is powered from a Class 1E Bus power source, but all four channels are powered by the same source. Since this is not in compliance with the Regulatory Guide, Rev. 3 recommendations, and based on the information given above, we are noting our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for S/G level instrumentation. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.16.)

2.2.7 Deviation No. DV-12; Condensate Storage Tank (CST) Level

CST level indication is currently provided in the control room through three Category 3 level-measuring devices. One of these instruments is electrically operated, while the other two are pneumatic devices. In addition, CST level can be read at the local turbine-driven auxiliary feedwater pump control panel. We have also committed to provide additional CST level indication by adding one instrument channel to meet Category 1 requirements. The new channel was installed during the 1987 refueling outage on Unit 1 and will be installed before the end of the 1988 refueling outage on Unit 2.

The CST is the initial source of water for the auxiliary feedwater (AFW) system, and provides sufficient volume to maintain the reactor

coolant system in a hot standby condition for nine hours. In the event that sufficient water is not available from the CST in one unit, operating procedures call for a cross-tie valve to be opened to supply feedwater from the CST in the other unit.

In the unlikely event that neither CST can supply sufficient AFW, procedures require transferring the supply source to the essential service water system (ESWS). The water supply for the ESWS is Lake Michigan.

The number and diversity of instrumentation available to provide CST level monitoring, and the ultimate availability of Lake Michigan as a source of auxiliary feedwater, is the basis for our deviation from the Regulatory Guide 1.97 recommendation to provide more than one Category 1 level indication for the CST. (This deviation was also discussed in AEP:NRC:0773S, Attachment 3, Item 1.)

2.2.8 Deviation No. DV-13; Containment Spray Flow

When operating normally, each containment spray pump will deliver 3200 gpm (design flow) at 490 ft TDH. Figure 2.2-1 shows containment spray pump flow as a function of pump discharge pressure.

The Figure 2.2-1 curve indicates the expected range of operation for the containment spray pumps. This operating range stems from consideration of pump suction head, containment pressure, and pump operating characteristics. Routine surveillance of spray pump operation is performed to ensure that, if containment spray is required, the pumps will operate in the indicated area of the flow curve and hence provide the necessary flow to the containment spray system. The reactor operators can, therefore, verify proper containment spray flow by monitoring spray pump discharge pressure to confirm that it is within the expected range.

It should be noted that the upper containment spray flow instrumentation (IFI-330 and IFI-331) cited in our original submittal (AEP:NRC:0773O, dated October 15, 1985) measures only the flow provided by the RHR pumps to the upper containment spray, not the flow from the containment spray pumps. However, the containment spray pumps, not the RHR pumps, are normally used to supply containment spray flow. Also, please note that the flow range of 0-200 gpm for IFI-330 and -331 (for measurement of RHR pump flow to the upper containment spray) given in that submittal is incorrect. The correct range is 0-2500 gpm.

The above is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for containment spray flow instrumentation. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.17.)

2.2.9 <u>Deviation No. DV-14; Volume Control Tank (VCT) Level</u>

Because of the following actions that apply for normal, accident, and post-accident conditions, we believe VCT level indication beyond that currently provided is not required. Upon receiving a hi-level alarm, flow into the VCT is automatically fully diverted into the hold-up tanks. If a low-level alarm is reached, an alarm alerts the operator to restore level. In the event this effort fails, an emergency lo-lo level alarm is sounded and the refueling water sequence is automatically initiated. We believe that this range (0-70 inches) is adequate to safely monitor the operation of this tank. In the unlikely event that VCT level indication is lost and the VCT becomes completely full, a safety relief valve (set at 75 psig) will open and the excess water will be discharged into the hold-up tanks. The above is the basis for our deviation from the Regulatory Guide 1.97 recommendations to monitor volume control tank level from top to bottom of the tank. (This deviation was also discussed in AEP:NRC:0773S, Attachment 1, Item 3.3.19.)

2.2.10 <u>Deviation No. DV-15; Noble Gases and Vent Flow from Condenser Air Removal System Exhaust</u>

This instrumentation was recently (1985) upgraded by the addition of a high-range noble gas detector. Based on our recent primary calibration analysis, the range of this instrumentation was determined to be 5.8×10^{-7} uCi/cc to 1.86×10^{4} uCi/cc Xenon-133 dose equivalent. On July 23, 1986, a letter was sent to the NRC (AEP:NRC:0678Y) in which we stated that post-accident conditions would not result in steam jet air ejector exhaust noble gas concentration greater than 2×10^{3} uCi/cc. On this basis we requested an exemption from the NUREG-0737, Section II.F.1-1 upper-range requirement 10^{5} uCi/cc in favor of a more realistic upper range of 10^{4} uCi/cc. The above also is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommended range for this parameter.

It should also be noted that tag numbers SFR-1900 and SFR-2900, given in our original submittal (AEP:NRC:0773O, dated October 15, 1985) are incorrect. The correct tag numbers are SRA-1900 and SRA-2900. (This information was also provided in AEP:NRC:0773S, Attachment 1, Item 3.3.21.)

2.2.11 <u>Deviation No. DV-16; Noble Gases from S/G Safety Relief Valves</u>

The range of 3 uCi/cc to 20×10^{5} uCi/cc as provided in our original submittal (AEP:NRC:0773O) was based solely on the monitor's response to Xe-133 and not to the anticipated mixture of radioisotopes following a steam generator tube rupture. The lower limit of 0.1 uCi/cc of XE-133 equivalent mixture can be measured. As stated in our September 8, 1986, letter (AEP:NRC:0678Z), when the anticipated mixture of radioisotopes for a steam generator tube rupture is used, the maximum concentration released in the main steam effluent from the S/G PORV is calculated to be 0.263 uCi/cc Xe-133 equivalent activity. With respect to this upper range limit, an exemption from the NUREG-0737 requirement of 1000 uCi/cc was requested in the September 8, 1986, letter and a 100 uCi/cc value

proposed. No response to our request has been received at this writing. The above is a basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommended upper limit of measurement for this variable. (This information was also provided in AEP:NRC:0773S, Attachment 1, Item 3.3.22.)

2.2.12 Deviation No. DV-17; RCS Sampling - Chloride Content

Chloride content in undiluted samples 30 days after an accident is measured in a range of 0.01 to 20 ppm. For diluted samples (1:1000 dilution) taken within 4 days of an accident, the range of measurement is 10 to 20,000 ppm. The above is the basis of our deviation from the Regulatory Guide 1.97, Rev. 3 lower limit of 0 ppm for this variable. (This information was also provided in AEP:NRC:0773S, Attachment 1, Item 3.3.24, No. 4.)

2.2.13 Deviation No. DV-18; Containment Air - Hydrogen Content

An exemption from the requirement for taking hydrogen grab samples of containment air was granted via a letter from Youngblood (NRC) to Dolan (AEP) dated November 5, 1986. This exemption is the basis for our deviation with respect to the Regulatory Guide 1.97, Rev. 3 recommendations for this parameter. We do, however, perform continuous monitoring of containment air hydrogen content in the range of 0 to 30 volume percent (see Item C-10).

2.2.14 Deviation No. DV-19; Containment Air - Oxygen Content

NUREG-0737 does not require sampling of containment air oxygen content. As noted above, however, we do continuously monitor hydrogen content, which makes containment air oxygen content of less concern from the standpoint of potential hydrogen flammability or deflagration. The above is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to sample for containment air oxygen content. (This deviation was also discussed in AEP:NRC:0773S, Item 3.3.24, No. 7.)

2.2.15 Deviation No. DV-20; Indicating Lamps

Circuits that require the use of indicating lamps for position indications, status indication, etc., will be using existing General Electric ET16 indicating lamps for this function. We have been advised by the manufacturer that these indicating lamps meet their (the manufacturer's) interpretation of IEEE-344-1975. This indicating lamp is a seismically rugged commercial grade device for which comprehensive qualification is not available. Since these lamps are purchased as standard commercial grade material and are not manufactured for a specific order, 10 CFR 21 can not be applied to these devices. The above is the basis for our deviation from Regulatory Guide 1.97, Rev. 3 for this device.

2.2.16 Deviation No. DV-21; Centrifugal Charging Pump (CCP) Flow Indication

Our original submittal (AEP:NRC:07730) identified both the CCP flow and CCP motor breaker status as Type A variables. This would require Category 1 instrumentation for monitoring these parameters. The CCP breaker status indication will meet the Regulatory Guide 1.97, Rev. 3 recommendations for Category 1 instrumentation except as noted in the summary table, Entry A.28.

With regard to the CCP flow indication, it should be noted that our Emergency Operating Procedures require manual operator action based on indication of pump operation or flow. The CCP breaker status indication and other parameters serve to verify pump operation. The non Category 1 CCP flow indication can serve as a backup. The above is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 instrumentation for CCP flow indication. (This deviation was also discussed in AEP:NRC:0773S, Attachment 3, Item 2).

2.2.17 Deviation No. DV-22; Safety Injection (SI) Pump Flow Indication

Reasons similar to those stated in Section 2.2.16 above are the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 instrumentation for SI pump flow indication. The SI pump motor breaker status instrumentation will meet Category 1 requirements except as noted in the summary table, Entry A.29. (This deviation was also discussed in AEP:NRC:0773S, Item 3.)

2.2.18 Deviation No. DV-23; Radiation Exposure Rate

ERA-7303 through 7308, and ERA-8303 through 8308 have ranges of 0.01 to 1000 R/HR. Monitors ERS-7401, 7403, 7404, 8401, and ERA-7507, 7601, 7603, and 7605 have ranges of .0001 to 10 R/HR. Monitors ERA-8403, 7504, 7508, 7602, and 7604 have ranges of 0.001 to 10 R/HR. These ranges are different from the Regulatory Guide 1.97, Rev. 3 recommended range of 10^{-1} to 10^{+4} R/HR. With the exception of ERA-7305, 7306, 8305, and 8306, the worst case maximum estimated accident dose rate in the areas where these monitors are to be installed is less than the upper range limits noted above. The stated upper range limits are used to provide more accurate, useful information and to help prevent false "low fail" alarms.

In the case of the ERA-7305, 7306, 8305, and 8306, the worst case maximum estimated accident dose rate is 1730 R/HR, which exceeds the detector's upper range limit of 1000 R/HR. However, within one (1) hour, this dose rate drops to 573 R/HR, which is well within the upper range limit. Personnel entry into an area where exposure may exceed 1000 R/HR (indicated by a "high fail" status indication) is highly unlikely and the dose rate in these areas will quickly fall below the upper range limit of 1000 R/HR (at which time a quantitative indication will again be available). Therefore, the range of 0.01 to 1000 R/HR for these detectors is adequate for the areas in which they are installed.

The above discussion is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommended range for area radiation monitors installed in areas potentially requiring personnel access for servicing of equipment important to safety.

2.2.19 Deviation No. DV-24; Valve Position Indication - QCM-250

We will upgrade the valve position limit switches on valves VCR-11 and VCR-21 to meet the environmental qualification requirements of 10 CFR 50.49 and Regulatory Guide 1.97, Rev. 3 recommendations.

The valve position limit switches for valve QCM-250, as well as the associated cable and terminations, are qualified in accordance with 10 CFR 50.49(k) except that they have not been qualified for submergence. The QCM-250 position indication limit switch is located below maximum flood level. Although this is not completely consistent with the Regulatory Guide 1.97, Rev. 3 recommendations for equipment qualification, we do not believe any upgrading of the position indication limit switch is necessary. This is due to the fact that QCM-250 is designed to close within 15 seconds of a containment isolation signal, which means that the valve will not become submerged before it performs its safety function. In addition, once the valve is closed, it is extremely unlikely that it would change position due to its submergence.

Given these considerations, we believe that QCM-250 in its present status, without upgrading, adequately meets the intent of Regulatory Guide 1.97, Rev. 3 recommendations for achieving verifiable containment isolation and is the basis for our deviation from the Regulatory Guide 1.97, Rev. 3 recommendations for this parameter.

The planned schedule for upgrading VCR-11 and VCR-21 to meet 10 CFR 50.49 requirements calls for this work to be completed in both units by the end of the refueling outages presently scheduled for 1989 (Unit 1) and 1990 (Unit 2).

2.2.20 Deviation No. DV-25; CCW Water Temperature Indication CTR -410, -415, -420 and -425

If CCW water temperature is not available, adequate CCW cooling can be verified by monitoring CCW flow and RHR inlet and outlet temperatures, all of which are qualified (or planned to be qualified) for the intended purpose. Therefore, because of the availability of suitable diverse indications, environmental qualification of instrumentation monitoring this variable is not required. We are submitting a deviation with respect to Reg. Guide 1.97, Rev. 3 recommendations for this variable for environmental qualification. (Deviation No. DV - 25)

2.3 SUMMARY OF REGULATORY GUIDE 1.97, REV. 3 DEVIATIONS

Table 2.3-1 provides a cross-referenced summary of the deviations from the Regulatory Guide 1.97, Rev. 3 recommendations that were identified and discussed in Sections 2.1 and 2.2. Table 2.3-1 shows, by deviation number, a brief description of each deviation, where the affected variables can be found in the Section 3.0 summary tables and where each deviation is discussed in Section 2.0.

2.4 ADDITIONAL CLARIFICATION OF SECTION 3.0 SUMMARY TABLE INFORMATION

The following information is provided in order to clarify certain entries contained in the Section 3.0 summary tables.

2.4.1 Sensor Location

The "sensor(s) location(s)" information requested in Section 6.2(e) of Supplement 1 to NUREG-0737 is assumed to mean the parameter(s) monitored by the sensor and not the sensor's physical plant location. This information is provided in the column labeled "variable."

2.4.2 Power Source

Instruments reported as conforming to the Regulatory Guide 1.97, Rev. 3 power source recommendations have their power derived from a 120V AC or 250V DC safety-related power source. Some existing circuits use non-safety related cable from the source to the instrument power supply.

2.4.3 Schedule

The implementation provided for each variable is the current best estimate of the completion of the final configuration for the associated instrument including redundancy, final displays, power supplies, documentation of qualification and turnover to plant operations. The schedules are based on anticipated delivery and plant outage schedules. Equipment delivery delays, environmental qualification test difficulties, or other problems, however, may cause delays in these schedules beyond our reasonable control. Considering these factors, our overall target date for completion of work undertaken in response to Regulatory Guide 1.97, Rev. 3 is by the end of the Unit 2 refueling outage presently scheduled for 1990. This completion date is also contingent on any changes resulting from the NRC evaluation of our deviations.

2.4.4 Changes from October 1985 Submittal (AEP:NRC:07730)

Because of additional information obtained after our previous submittal in October 1985, discussions with the NRC Staff and its consultant (EG&G), and information developed as a result of our ongoing engineering/design work, there are numerous differences between our October 1985 submittal and this updated submittal. All item numbers in the tables in Section 3.0 have been retained, regardless of whether they are currently in use, in order to provide easy cross referencing between this and previous reports. Most changes involve the updating of status of modifications, incorporation of deviation requests and answers to questions

provided in our AEP:NRC:0773S documents, clarification of several items, and miscellaneous correction and revisions.

It is noted that although many variables presently listed are measured with fully functional instrumentation, they are not considered operational in the sense that they meet Regulatory Guide 1.97, Rev. 3 recommendations. In addition, it should be noted that instruments previously reported to the staff as meeting the requirements of NUREG-0737 or IE Bulletin 79-01B do not, in all cases, meet all of the requirements of Regulatory Guide 1.97, Rev. 3 as noted within this document.

3.0 SUMMARY TABLES

These tables represent the status of the Donald C. Cook Nuclear Plant when we will have completed all of the recommendations associated with Regulatory Guide 1.97, Rev. 3. It does not, nor is it intended to, reflect the status of the plant as of the date of this letter.

The schedules provided in this enclosure for meeting Regulatory Guide 1.97, Rev. 3 recommendations are not intended to change any previous commitment regarding NUREG-0737 or 10 CFR 50.49, whose requirements may be different.

"A:" A letter "A" in the columns labeled EQ, SQ, QA, SF or PS indicates that we have applied the guidance provided in Regulatory Guide 1.97, Rev. 3 subject to specific deviations noted in the Remarks/Action Req'd. column.

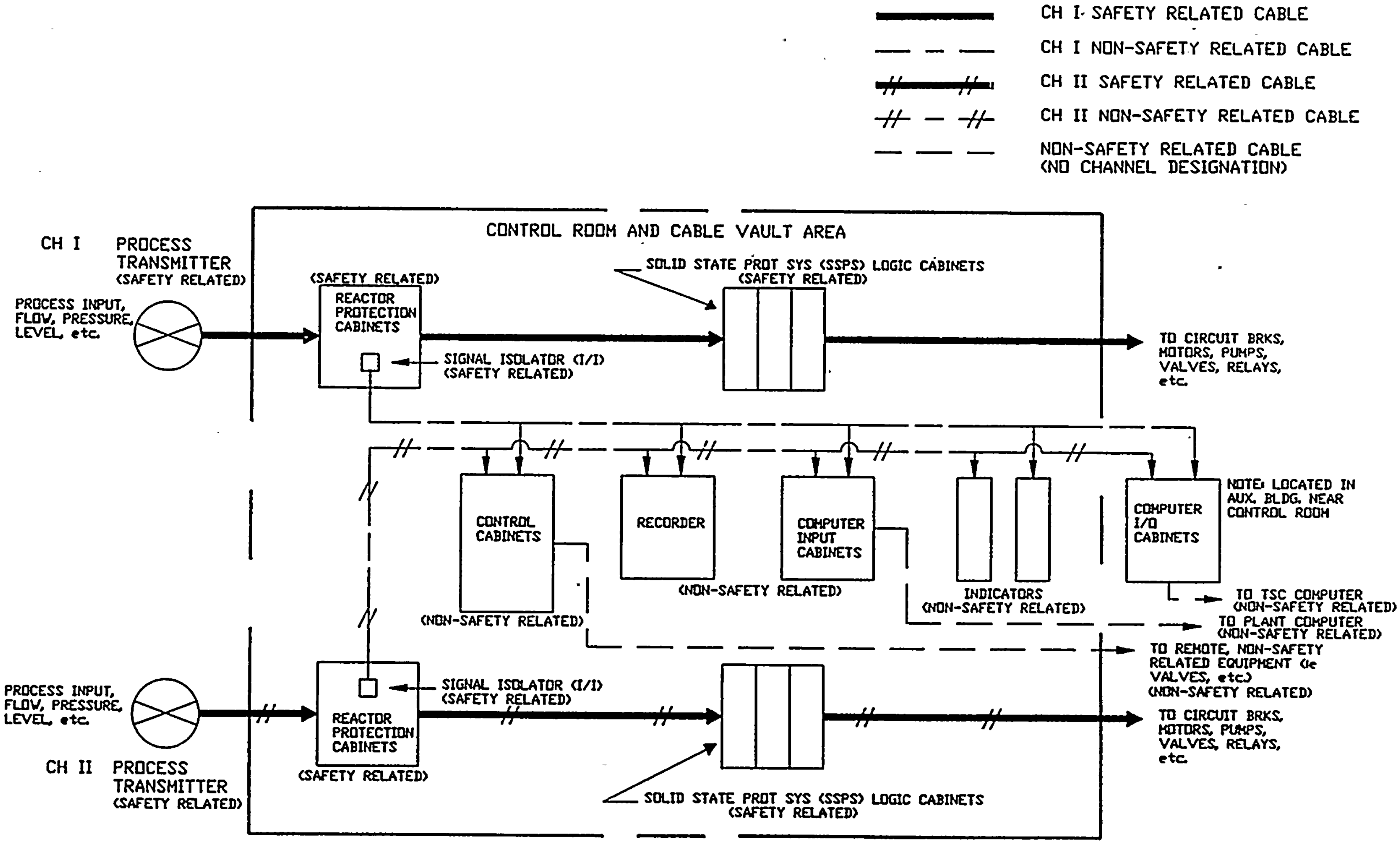
"CMPLT" indicates that the action identified in our previous submittal AEP:NRC:07730 has been completed on Unit 1 or will be completed on Unit 2 prior to returning to service from the current steam generator repair and refueling outage.

"NA" letters mean Not Applicable; i.e., does not apply.

TABLE 2.3-1

Summary of Deviation Requests from Regulatory Guide 1.97, Rev. 3

Deviation No.	Description	Where Identified In Section 3.0	Where Explained
DV-1	Environmental Qualification	Various	Section 2.1.1
DV-2	Seismic Qualification	Various	Section 2.1.2
DV-3	Quality Assurance	Various	Section 2.1.3
DV-4	Redundancy	Various	Section 2.1.4
DV-5	Display Location	Various	Section 2.1.5
DV-6	Containment Pressure	Table Items A.13/B.13	Section 2.2.1
DV-7	Subcooling Meter	Table Item A.17	Section 2.2.2
DV-8	RCS Sampling/Boron Concentration Range	Table Item B.3	Section 2.2.3
DV-9	RCS Sampling/Radioactivity	Table Item C.2	Section 2.2.4
DV-10	Quench Tank Level	Table Item D.13	Section 2.2.5
DV-11	S/G Wide Range Level	Table Item D.15	Section 2.2.6
DV-12	CST Level	Table Item D.19	Section 2.2.7
DV-13	Containment Spray Flow	Table Item D.20	Section 2.2.8
DV-14	VCT Level	Table Item D.26	Section 2.2.9
DV-15	Noble Gases and Vent Flow from Condenser Air Removal System Exhaust	Table Item E.3d	Section 2.2.10
DV-16	Noble Gases from S/G Safety Relief Valves	Table Item E.3f	Section 2.2.11
DV-17	RCS Sampling - Chloride Content	Table Item E.9d	Section 2.2.12
DV-18	Containment Air - H_2 Content	Table Item E.10a	Section 2.2.13
DV-19	Containment Air - O_2 Content	Table Item E.10b	Section 2.2.14
DV-20	Indicating Lamps	Table Item A.28, A.29 and B.14	Section 2.2.15
DV-21	Centrifugal Charging Pump Flow Indication	Table Item A.01	Section 2.2.16
DV-22	SI Pump Flow Indication	Table Item A.34	Section 2.2.17
DV-23	Radiation Exposure Rate	Table Item E.2	Section 2.2.18
DV-24	Position Indication - QCM-250	Table Item B.14	Section 2.2.19
DV-25	CCW Water Temperature	Table Item D.27	Section 2.2.20



TYPICAL POST ACCIDENT MONITORING SIGNAL CABLING & HARDWARE LAYOUT
FIGURE 2.1-1

CONTAINMENT SPRAY PUMP FLOW VS. DISCHARGE PRESSURE

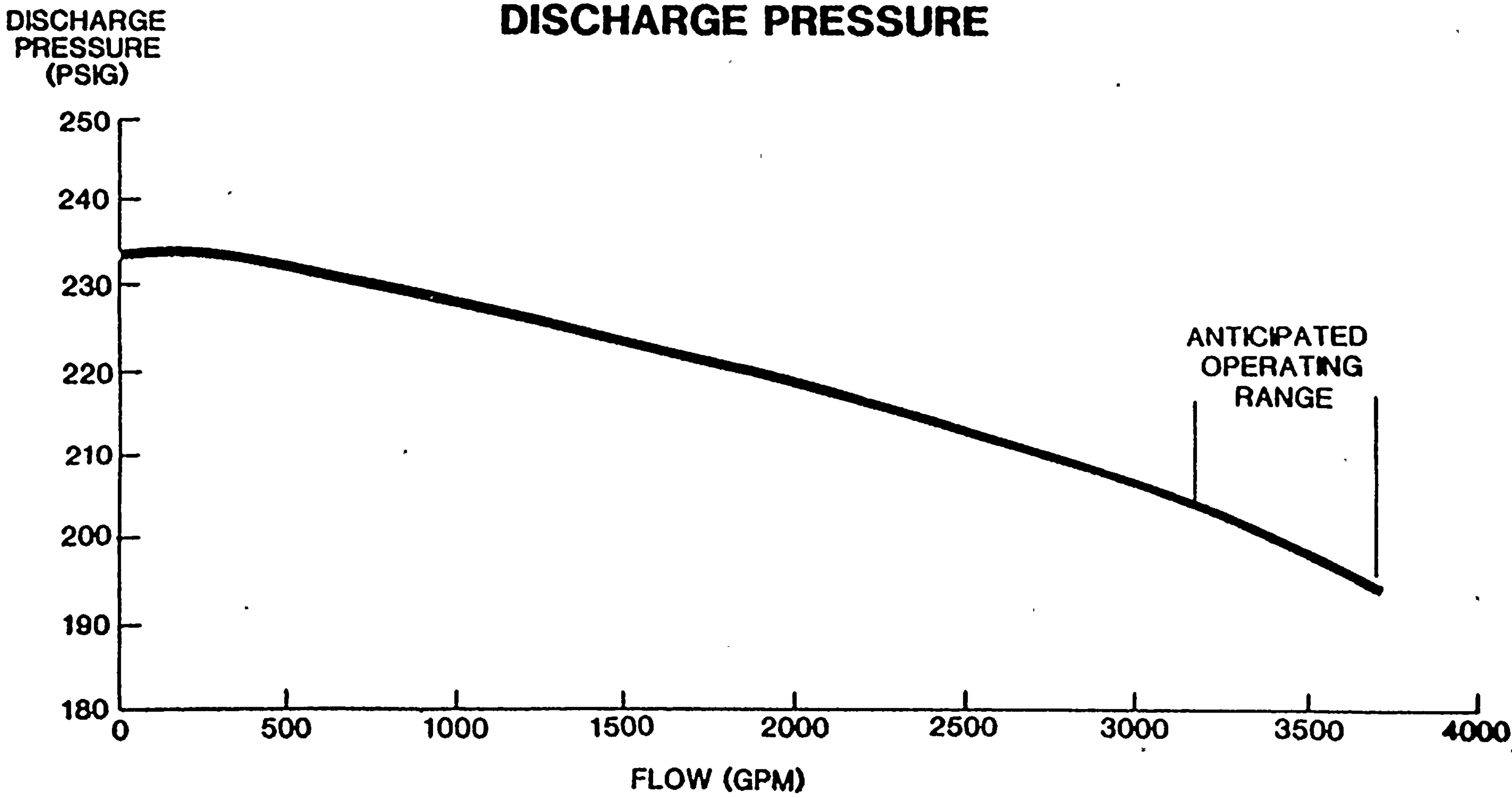


FIGURE 2.2-1

Type A Variables: "those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. Primary information is information that is essential for the direct accomplishment of the specified safety funtions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures." Note: These variables are plant-specific and based on review of the D. C. Cook Nuclear Plant Emergency Operating Procedures (EOP's) plus anticipated future changes to the EOP's. The schedule and status for each instrument indicated is for when all of the applicable recommendations of Regulatory Guide 1.97, Rev 3 will be met.

Item No.	Purpose	Variable	Tag Nos.	Range	EQ	SQ	QA	SF	PS	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
A-1	Maintain Pressurizer Level during S/G Tube rupture	Centrifigal Chg Pump Flow (CCP)	IFI-51,52, 53,54	0-200 GPM	A	A	A	A	A	Control Room Panel SIS	No further action required See footnote (t) (Deviation No.s DV-21 and DV-5)	CMPLT	CMPLT
A-2	Manual Trip of RC Pumps based on RCS pressure	RCS Pressure (wide range)	NPS-121,122	0-3000psig	A	A	A	A	A	Control Room Panel RHR	No further action required (Deviation No.s DV-1, DV-2, DV-3, and DV-4)	CMPLT	CMPLT
A-3				NOT USED									
A-4				NOT USED									
A-5				NOT USED									
A-6				NOT USED									
A-7	Determination of required core exit temperature by S/G Pressure	S/G Pressure	MPP-210,211, 212,220, 221,222, 230,231, 232,240, 241,242	0-1200psig	A	A	A	A	A	Control Room Panel SG	No further action required (Deviation Nos. DV-1, DV-2, DV-3, DV-4)	CMPLT	CMPLT

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
A-8	Determination of adverse containment	Containment Water Level	NLA-320 NLI-321	599'-3" to 614 ft. elevation (Containment Floor to max flood level)	A	A	A	A	A	Control Room Panel RHR	Replace transmitters. See footnote (u) (Deviation Nos. DV-1, DV-2, DV-3, & DV-5)	1989 refueling outage	CMPLT
A-9	Manual Reduction of ECCS Flow (Secondary heat sink capability)	S/G Level Narrow range	BLP-110,111, 112,120, 121,122, 130,131, 132,140, 141,142	From below 1st stage separator to 2nd stage separator	A	A	A	A	A	Control Room Panel SG	No further action required (Deviation No.s DV-1, DV-2, DV-3, DV-4)	CMPLT	CMPLT
A-10	Manual Reduction of ECCS Flow	Pressurizer Level	NLP-151,152, 153	0-100% (96% of Total Volume)	A	A	A	A	A	Control Room Panel PZR	No further action required (Deviation No.s DV-1, DV-2, DV-3, and DV-4)	CMPLT	CMPLT
A-11			NOT USED										
A-12	Determination of adverse containment	Containment Area Radiation Monitor High Range	VRA-1310, 1410, (Unit 1) 2310, 2410 (Unit 2)	1 R/HR to 1X10^7 R/HR	A	A	A	A	A	Control Room Panel RMS	See footnote (v) (Deviation Nos. DV-1, DV-2, DV-3, and DV-5)	NA	NA
A-13	Manually establish or trip containment spray	Containment Pressure (Narrow range)	PPP-300,301, 302,303	-5 to +12 psig	A	A	A	A	A	Control Room Panel SPY	No further action required See Footnotes (d) & (w) (Deviation No. DV-6)	CMPLT	CMPLT
A-14				NOT USED									

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Tag Nos.	Range	Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
A-15	Manual Reduction of ECCS Flow (Secondary heat sink capability)	Auxiliary Feedwater Flow	FFI-210,220, 230,240	0-250 $\times 10^3$ PPH	A	A	A	A	A	Control Room Panel SG	No further action required Redundancy provided by diverse variable - S/G narrow range level which is qualified (Deviation Nos. DV-1, DV-2, DV-3, DV-4, & DV-5)	CMPLT	CMPLT
A-16	Manual Transfer to cold leg recirculation in low level in RWST	RWST Level	ILS-950,951	essentially Top (bottom of overflow) to Bottom (100% of Total Volume)	NA	A	A	A	A	Control Room Panel SPY	No further action required (Deviation No.s DV-2, DV-3, & DV-4)	CMPLT	CMPLT
A-17	Manual Trip or reduction of Pressurizer Spray and ECCS Flow	Degrees Subcooling	NA	0-199°F subcooling 0-199°F Superheat	A	NA	A	NA	A	Control Room Panel BA	No further action required See footnotes (b) & (c) which apply beyond the isolating devices. Also see footnote (x) (Deviation No.s DV-7, DV-1, DV-3, & DV-5)	CMPLT	CMPLT
A-18				NOT USED									
A-19				NOT USED									
A-20				NOT USED									
A-21				NOT USED									
A-22				NOT USED									
A-23				NOT USED									
A-24				NOT USED									

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
A-25	Manual Reduction of ECCS Flow	Core Exit T/C's	T/C 1-65	200-2300°F	A	A	A	A	A	Control Room Panel FI (U-1) Panel RMS (U-2)	No further action required (Deviation No.s DV-1, DV-2, & DV-3)	CMPLT	CMPLT
A-26				NOT USED									
A-27				NOT USED									
A-28	Manual trip of RCP's	CCP Breaker Status	Pump IE,IW 2E,2W	OPEN/CLOSE	NA	A	A	A	A	Control Room Panel BA	Qualify or replace control room indicators with seismically qualified equipment See footnotes (d),(t) & (jjj) (Deviation No.s DV-2, DV-3, DV-20 & DV-21)	End of 1988	End of 1988
A-29	Manual trip of RCP's	SI Pump Breaker Status	Pump 1N,1S 2N,2S	OPEN/CLOSE	NA	A	A	A	A	Control Room Panel SIS	Qualify or replace control room indicators with seismically qualified equipment See footnotes (d), (y) & (jjj) (Deviation No.s DV-2, DV-3, DV-20 & DV-22)	End of 1988	End of 1988
A-30				NOT USED									
A-31				NOT USED									
A-32				NOT USED									
A-33				NOT USED									
A-34	Manual Trip of RCP's	Safety Injection Pump Flow	IFI-260,266	0-800GPM	NA	A	A	A	A	Control Room Panel SIS	No further action required See footnote (d) & (y) (Deviation No. DV-22 & DV-5)	CMPLT	CMPLT

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Tag Nos.	Range	Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
A-35		S/G Blowdown Radiation		NOT USED - DELETED							See footnote (z)	NA	NA
A-36				NOT USED									
A-37				NOT USED									

For Definition of "A" See Section 3.0

Type B Variables: "those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control)." Note: The schedule and status of each instrument is for when all of the applicable recommendations of Regulatory Guide 1.97, Rev 3 will be met.

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
B-1	Reactivity Control	Neutron Flux	1	NE-21,23	10^{-9}-200% power	A	A	A	A	A	Control Room Panel NIS	Install New, Cat. 1 Channel to provide Indication. Upgrade 1 existing channel to meet Category 1 requirements. See footnote (aa) (Deviations No.s DV-1, DV-2 & DV-3)	1989 refueling outage	1990 refueling outage
B-2		Control Rod Position	3	CA1-8, CB1-4 CC1-8, CD1-9 SA1-8, SB1-8	Full in or not full in	NA	NA	NA	NA	NA	Control Room Panel RC	None required	NA	NA
B-3		RCS Soluble. Boron Concentrate	3	NSX-101,103	375-2000 ppm	NA	NA	NA	NA	NA	NA	See Item C-2 and footnotes (a) & (bb) (Deviation No. DV-8)	NA	NA
B-4		RCS Cold Leg Temperature	1	NTR-210,230,	0-700°F	A	A	A	A	A	Control Room Panel DTU	Relocate R/I and I/I converters to control room. Upgrade cable to meet Category 1 requirements. See footnote (cc) (Deviation No.s DV-1, DV-2, DV-3 & DV-4)	1989 refueling outage	1990 refueling outage
B-5	Core Cooling	RCS Hot Leg Water Temperature	1	NTR-110,130,	0-700°F	A	A	A	A	A	Control Room Panel DTU	Relocate R/I and I/I converters to control room. Upgrade cable to meet Category 1 requirements. See footnote (cc) (Deviation No.s DV-1, DV-2, DV-3 & DV-4)	1989 refueling outage	1990 refueling outage

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
B-6		RCS Cold Leg Water Temperature	1									See item B-4		
B-7		RCS Pressure										See item A-2		
B-8		Core Exit Temperature										See item A-25		
B-9		Coolant Inventory	1	NLI-110,111 120,121, 130,131	Top of head vent piping to bottom of vessel (100% of Volume)	A	A	A	A	A	Control Room Panel SIS	Upgrade Power Supply to Hot Leg Temperature Input. See Item B-4 which provides an input to this variable (Deviation No.s DV-1, DV-2, DV-3 & DV-5)	1989 refueling outage	1990 refueling outage
B-10		Degrees of Subcooling										See item A-17		
B-11	Maintaining RCS Integrity	RCS Pressure										See item A-2		
B-12		Containment Sump Water Level	2	NLA-310 NLI-311	589'-6" to 599'-8" (Bottom of Sump to Containment Floor)	A	NA	A	NA	NA	Control Room Panel RHR	See footnote (u) & (dd) (Deviation No.s DV-1 & DV-3)	1989 refueling outage	CMPLT
B-13		Containment Pressure (Wide Range)	1	PPA-310,312	-5 to 36psig	NA	A	A	A	A	Control Room Panel SPY	No further action required. Also see Item A-13 for narrow range. See footnotes (d) and (w) narrow range (Deviation No. DV-6)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	EQ	SQ	QA	SF	PS	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
B-14	Maintaining Containment Integrity	Containment Isolation Valve Position (excluding check valves)	1	See listing in Attachment 1 to these tables	CLOSED - NOT CLOSED	A	A	A	A	A	Control Room Panels IV, BA,SIS,SPY	Replace Limit switches with env & seismically qualified devices as noted in Attachments 1 & 2 to these tables. Qualify or replace control room indicators with seismically qualified equipment. See footnote (ee) & (jjj) (Deviation No.s DV-20 DV-1, DV-2 & DV-3)	1989 refueling outage	1990 refueling outage
B-15		Containment Pressure										See items A-13 and B-13		

For Definition of "A" See Section 3.0

Type C Variables: "those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment."
Note: The schedule and status for each instrument is for when all of the applicable recommendations of Regulatory Guide 1.97, Rev 3 will be met.

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
C-1	Fuel Cladding	Core Exit Temperature										See item A-25		
C-2		Radioactive concentration or Radiation Level in Circulating Primary Coolant	1	NSX-101,103	NA	A	A	A	A	A	NA	No further action required. See footnotes (h) and (ff) (Deviation No.s DV-9, DV-1, DV-2, DV-3 & DV-5)	CMPLT	CMPLT
C-3		Analysis of Primary Coolant (Gamma Spectrum)										See item C-2 and footnote (gg)		
C-4	Reactor Coolant Pressure Boundary	RCS Pressure										See item A-2		
C-5		Containment Pressure										See items A-13 and B-13		
C-6		Containment Sump Water Level										See item B-12		
C-7		Containment Area Radiation										See item A-12		

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
C-8		Effluent Radioactiv- ity-Noble gas Effluent from Con- denser Air Removal Sys- tem Exhaust	3	SRA-1900 (Unit 1 SRA-2900 (Unit 2)	5.8E-7 to 1.86E4 uCi/cc	NA	NA	NA	NA	NA	Control Room CT-1 Control Terminal	No action required	NA	NA
C-9	Containment	RCS Pressure										See item A-2		
C-10		Containment Hydrogen Concentra- tion	1	ESR-1 thru 9	0-30 Volume %	NA	A	A	A	A	Control Room Panel IV	See footnote (d) No action required (Deviation No.s DV-1, DV-2, & DV-3)	NA	NA
C-11		Containment Pressure										See items A-13 and B-13		
C-12		Containment Effluent Radioactiv- ity-Noble gases from identified release points	2	VRS-1500, (Unit 1) VRS-2500, (Unit 2)	5.8E-07 to 1.86E4 uCi/cc	NA	NA	A	NA	NA	Control Room CT-1 Terminal	See footnotes (d) & (hh) (Deviation No. DV-3) No Action Required	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	EQ	SQ	QA	SF	PS	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
C-13		Effluent Radioactivity-Noble Gases (from buildings or areas where penetrations and hatches are located, eg, secondary containment and AUX buildings that are in direct contact with primary containment										SEE ITEM C-12		

For Definition of "A" See Section 3.0

Type D Variables: "those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident." Note: The schedule and status for each instrument is for when all of the applicable recommendations of Regulatory Guide 1.97, Rev 3 will be met.

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	EQ	SQ	QA	SF	PS	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-1	RHR System	RHR System Flow	2	IFI-310,311, 320,321	0-1500 GPM 1500-5000 GPM	A	NA	A	NA	NA	Control Room Panel RHR	No Action Required See footnotes (jj) & (mmm) (Deviation No. DV-3)	NA	NA
D-2		RHR Heat Exchange Outlet Temp	2	ITI-310,320	0-400°F	A	NA	A	NA	NA	Control Room Panel RHR	No further action required (See footnote (ii) (Deviation No.s DV-1, & DV-3)	CMPLT	CMPLT
D-3a	SI Systems	Accumulator Tank Level	2	ILA-110,111 120,121, 130,131, 140,141	4.148 to 120.8 in. (wide range) (52% of Total Volume) 104.15 to 120.8 in. (narrow range) (7.5% of Total Volume)	A	NA	A	NA	NA	Control Room Panel SIS	Replace the wide range transmitter each tank with env. qual. equipment See footnotes (iii) Qualification shall apply to the wide range instruments only. (Deviation No.s DV-1, & (DV-3)	1989 refueling outage	1990 refueling outage
D-3b		Accumulator Tank Pressure	2	IPA-110,111, 120,121, 130,131, 140,141	0-800 psig	A	NA	A	NA	NA	Control Room Panel SIS	Replace one transmitter/tank with env. qualified equipment. Qualification shall apply to only one instrument/tank. (Deviation No.s DV-1, & DV-3)	1989 refueling outage	1990 refueling outage
D-4		Accumulator Tank Isolation Valve Position	2	IMO-110,120, 130,140	Closed or Open	NA	NA	A	NA	NA	Control Room Panel SIS	None Required See footnote (r) & (kk) (Deviation No. 3)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-5		Boric Acid Charging Flow										See item D-24		
D-6		Flow in HPI System										See item A-1		
D-7		Flow in LPI System										See item D-1		
D-8		RWST Level										See item A-16		
D-9	Primary Cool-ant System	RCP Status	3	Q1,Q2,Q3,Q4	0-1200A	NA	NA	NA	NA	NA	Control Room Panel RCP	No Action Required	NA	NA
D-10		Primary Sys-tem Safety Relief Valve Positions or Flow Thru or pressure in Relief Valve Lines	2	QR-107 A,B, C,D	NA	A	NA	A	NA	NA	Control Room Panel RC	See footnote (b) No Action Required (Deviation No.s DV-1, DV-3)	NA	NA
D-11		Pressurizer Level										See item A-10		
D-12		Pressurizer Heater Sta-tus	2	Group A1,A2, A3,C1,C2,C3	ON/OFF	NA	NA	A	NA	NA	Control Room Panel PZR	See footnotes (d) & (11) No Action Required (Deviation No. DV-3)	NA	NA
D-13		Quench Tank Level	3	NLA-351	7 inches above tank bottom to 7 inches below tank top	NA	NA	NA	NA	NA	Control Room Panel PZR	None required See footnote (mm) (Deviation No. DV-10)	NA	NA
D-14		Quench Tank Temperature	3	NTA-351	50-750°F	NA	NA	NA	NA	NA	Control Room Panel PRZ	No further action required	CMPLT	CMPLT

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S.	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-14a		Quench Tank Pressure	3	NPA-351	-10 to 100 psig	NA	NA	NA	NA	NA	Control Room Panel PRZ	None Required	NA	NA
D-15	Secondary System (Steam Generator)	S/G Level	1	BLI-110,120, 130,140 (wide range)	From 12" above tube sheet to separators	A	A	A	A	A	Control Room Panel SG	None required See footnotes (i) & (oo) (Deviation No. DV-11)	NA	NA
D-16a		Safety/Relief Valve Positions	2									See Item D-16b for alternate instrumentation		
D-16b		Main Steam Flow	2	MFC-110,111, 120,121, 130,131, 140,141	0-4x10^6 PPH	A	NA	A	NA	NA	Control Room Panel SG	None required (Deviation No.s DV-1, DV-3)	NA	NA
D-17		Main Feedwater Flow	3	FFC-210,211, 220,221, 230,231, 240,241	0-4x10^6 PPH	NA	NA	NA	NA	NA	Control Room Panel BA	None required See footnote (pp)	NA	NA
D-18	Auxiliary Feedwater System	Aux Feedwater Flow										See item A-15		
D-19		Condensate Storage Tank Level	1	CLI-113,114 CLR-110, 111	Essentially Top to Bottom (95% Total Volume)	A	A	A	A	A	Control Room Panel CP	No further action required. See footnotes (qq) & (mmm) (2 channels only) (Deviation No. DV-12)	CMPLT	CMPLT
D-20	Containment Cooling System	Containment Spray Flow	2	IFI-330,331 (Upper containment)	0-2500GPM	NA	NA	A	NA	NA	Control Room Panel SPY	No Action Required See footnotes (d), (e), and (rr) (Deviation No.s DV-13, DV-3)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S.	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-21		Heat Removal by containment Heat Removal System										D. C. Cook Nuclear Plant Units 1&2 do not have a Containment Heat Removal System, therefore this item does not apply	NA	NA
D-22		Containment Atmosphere Temperature	2	ETR-11,12,13, 14,15,16, 17,18,19, 20,21,22; 23,24,25, 26,27	0 to 400°F	A	NA	A	NA	NA	Control Room Panel A-14	Replace six (6) tech spec related RTD's with environmentally qualified equipment. Increase range from -0 to 300°F to as specified. Qualification shall apply for the six (6) T.S. related instr. only. (Deviation No.s DV-1, DV-3)	1989 refueling outage	1990 refueling outage
D-23		Containment Sump Water Temperature	2	ITR-311,321	50 to 400°F	A	NA	A	NA	NA	Control Room Panel RHR	Replace RTD's with qualified equipment. See footnote (ss) (Deviation No.s DV-1, DV-3)	1989 refueling outage	1990 refueling outage
D-24	Chemical and Volume Control System	Make up Flow-In	2	QFI-200	0-200 GPM	NA	NA	A	NA	NA	Control Room Panel BA	No Action Required See footnote (d) (Deviation No. DV-3)	NA	NA
D-25		Letdown Flow Out	2	QFI-301	0-200 GPM	NA	NA	A	NA	NA	Control Room Panel BA and HSD	No Action Required See footnote (d) (Deviation DV-3)	NA	NA
D-26		Volume Control Tank Level	2	QLC-451,452	Essentially top to bottom (65% of Total Volume)	NA	NA	A	NA	NA	Control Room Panel BA	No Action Required See footnote (d), and (tt) (Deviation No.s DV-14 & DV-3)	NA	NA
D-27	Cooling Water System	CCW water Temperature to ESF System	2	CTR-410,415, 420,425	0-200°F	A	NA	A	NA	NA	Control Room Panel ESW	No Action Required See footnote (d) & (111) (Deviation No. DV-3 & DV-25)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S.	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-28		CCW Flow to ESF System	2	CFI-410,419, 420,429	0-10000GPM 0-6000GPM	A	NA	A	NA	NA	Control Room Panel CCW	No Action Required See footnote (mmm) (Deviation DV-3)	NA	NA
D-29	Radwaste Systems	High Level Radioactive Liquid Tank Level	3	RLS-255,256	Essentially Top to Bottom (84% of Total Volume)	NA	NA	NA	NA	NA	Panel WDG	None required	NA	NA
D-30		Radioactive Gas Holdup Tank Pressure	3	RPC-310,320, 330,340, 350,360, 370,380	0-225 psig	NA	NA	NA	NA	NA	Panel WDG	No further action required	CMPLT	CMPLT
D-31	Ventilation System	Emergency Ventilation Damper Position	2	VCR-201 thru 207	Open-Closed	A	NA	A	NA	NA	Control Room Panel IV VCR-207 on Control Room Panel SPY	Replace Limit Switches VCR-201 & 202 with env. qualified equipment. Footnote (d) applies to VCR-203 thru 207. See footnote (s) (Deviation Nos. DV-1, DV-3)	1989 refueling outage	1990 refueling outage
	Power Supplies	Status of Standby Power and other Energy Sources Important to Safety												
D-32a		Diesel Gen Status	2	DG1AB DG1CD	0-800A	NA	NA	A	NA	NA	Control Room Panel SA	See footnote (d) No Action Required (Deviation No. DV-3)	NA	NA
D-32b		4KV Safety Related Po- Power Systems Status	2	Bus T11A, T11B, T11C T11D	0-150V	NA	NA	A	NA	NA	Control Room Panel SA	See footnote (d) No Action Required (Deviation No. DV-3)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
D-32c		250VDC Battery Power System Status	2	Battery AB Battery CD	0-300V	NA	NA	A	NA	NA	Control Room Panel SA	See footnote (d) No Action Required (Deviation No. DV-3)	NA	NA
D-32d		120VAC Safety Related Power System Status	2	Channel I,II, III, IV	0-150V	NA	NA	A	NA	NA	Control Room Panel SA	See footnote (d) No Action Required (Deviation No. DV-3)	NA	NA
D-32e		Instrument Air Status	2	XPI-100 XPI-50 XPI-20 XPI-85	0-150psig 0-100psig 0-60psig 0-160psig	NA	NA	A	NA	NA	Control Room Panel SV	These are mechanical devices - no electrical components No Action Required (Deviation No. DV-3)	NA	NA

For Definition of "A" See Section 3.0

Type E Variables: "those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases." Note: The schedule and status for each instrument is for when all of the applicable recommendations of Regulatory Guide 1.97, Rev. 3 will be met.

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
E-1	Containment Radiation	Containment Area Radiation High Range										See item A-12		
E-2	Area Radiation	Radiation Exposure Rate (inside buildings or where areas of access are required to service equipment important to safety)	3	ERA-7303 thru 7308 ERA-8303 thru 8308 ERS-7401 ERA-7403, 7404 ERS-8401 ERA-8403, ERA-7504, 7507, 7508 ERA-7601 thru 7605	See Footnote (kkk)	NA	NA	NA	NA	NA	Control Room CRT	Install new monitors See footnote (uu) & (kkk) (Deviation No. DV-23)	By end of 1989	By end of 1989
E-3a	Noble Gases and vent Flow Rate	Containment or Purge Effluent	2		SEE ITEM E-3e									
E-3b		Reactor Shield Building Annulus	2		SEE ITEM E-3e									
E-3c		Aux Building	2		SEE ITEM E-3e									
E-3d		Condenser Air Removal System Exhaust	2	SRA-1900 (Unit 1) SRA-2900 (Unit 2) SFR-401	5.8E-07 to 1.86E4 uCi/cc 0-250 scfm	NA	NA	A	NA	NA		See footnotes (d), (ww) and (xx) (Deviation No.s DV-15 & DV-3)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
E-3e		Common Plant Vent	2	VRS-1500 (Unit 1) VRS-2500 (Unit 2) VRF-315	5.8E-07 to 1.86E4 uCi/cc 0-200K scfm	NA	NA	A	NA	NA	Control Room CT-1 Control Terminal	None Required See footnote (d) (Deviation DV-3)	NA	NA
E-3f		Vent from S/G Safety Relief Valves	2	MRA-1600 1700 (Unit 1) MRA-2600 2700 (Unit 2)	0.01 to 100 uCi/cc	NA	NA	A	NA	NA	Control Room Panel RMS	None Required See footnote (yy) (Deviation No.s DV-16 & DV-3)	NA	NA
E-3g		Other identified release points	2	SRA-1800 (Unit 1) SRA-2800 (Unit 2) SFR-201	5.8E-07 to 1.86E4 uCi/cc 0-1000 scfm	NA	NA	A	NA	NA	Control Room Panel FI	None Required See footnote (d) (Deviation No. DV-3)	NA	NA
E-4	Particulates and Halogens	All identified release points (except S/G Safety Relief valves and Condenser air removal System exhaust) Sampling and onsite analysis	3									See Item E-3e		
E-5a	Environ Radiation and Radioactivity	Airborne Radioactivity and Particulates sampling and analysis (portable)	3	NA	1E-9 to 1E-3 uCi/cc (minimum)	NA	NA	NA	NA	NA	NA	None Required	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	E Q	S Q	Q A	S F	P S.	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
E-5b		Plant and Environs Radiation (Portable)	3	NA	Gamma 1.0E-3 to 1.0E4 R/hr. Beta/low energy gama 1.0E-3 to 1.0E4 Rad/hr	NA	NA	NA	NA	NA	NA	Completion of Calibration	End of 89	End of 89
E-5c		Plant and Environs Radioacti-vity (Port-able)	3	NA	Isotopic Analysis	NA	NA	NA	NA	NA	NA	No further action required. See footnote (zz)	CMPLT	CMPLT
E-6	Meteorology	Wind Direc-tion	3	EFR-410,412, 413,414	0-360°	NA	NA	NA	NA	NA	Control Room Panel Flx and/or CRT	None required	NA	NA
E-7		Wind Speed	3	EFR-400,404, 402,403	0-100 mph	NA	NA	NA	NA	NA	Control Room Panel Flx and/or CRT	None required	NA	NA
E-8		Estimation of Atmos-pheric Stability	3	ETR-400,402, 403 ETQ-401	-30 to 50°C	NA	NA	NA	NA	NA	Control Room Panel Flx and/or CRT	None required	NA	NA
E-9a	Accident Samp-ling Primary Coolant and Sump	Gross Acti-vity	3	NSX-101,103 ESX-400	1 uCi/ml to 10 Ci/ml	NA	NA	NA	NA	NA	NA	See Item C-2 See footnote (aaa)	NA	NA
E-9b		Gamma Spec-trum	3	NSX-101,103 ESX-400	0.050 to 2.05 MeV Isotopic Analysis	NA	NA	NA	NA	NA	NA	See item C-2 See footnote (bbb)	NA	NA

For Definition of "A" See Section 3.0

Item No.	Purpose	Variable	Cat.	Tag Nos.	Range	EQ	SQ	QA	SF	PS	Display Location	Remarks/Action Req'd	U-1 Schedule	U-2 Schedule
E-9c		Boron Content	3	NSX-101,103 ESX-400	375 to 2000 ppm	NA	NA	NA	NA	NA	NA	See Item C-2 See footnote (ccc) & (bb)	NA	NA
E-9d		Chloride Content	3	NSX-101,103 ESX-400	0.01 to 20 ppm	NA	NA	NA	NA	NA	NA	SEE ITEM C-2 (See footnote (ddd) (Deviation No. DV-17)	NA	NA
E-9f		Dissolved H_2 or total gas	3	NSX-101,103 ESX-400	0-2000 cc/kg	NA	NA	NA	NA	NA	NA	SEE ITEM C-2	NA	NA
E-9g		Dissolved O_2	3	NSX-101,103 ESX-400	0-20 ppm	NA	NA	NA	NA	NA	NA	SEE ITEM C-2	NA	NA
E-9h		pH	3	NSX-101,103 ESX-400	1.0 to 13.0 pH	NA	NA	NA	NA	NA	NA	SEE ITEM C-2 See footnote (eee)	NA	NA
E-10a	Containment Air	H_2 Content,	3	ESX-001	NA	NA	NA	NA	NA	NA	NA	None required See footnote (fff) (Deviation No. DV-18)	NA	NA
E-10b		O_2 Content	3	NA	NA	NA	NA	NA	NA	NA	NA	None required See footnotes (q) and (ggg) (Deviation No. DV-19)	NA	NA
E-10c		Gamma Spectrum	3	ESX-001	1 uCi/cc to 10 Ci/cc Isotopic Analysis	NA	NA	NA	NA	NA	NA	None required See footnote (hhh)	NA	NA

For Definition of "A" See Section 3.0

ATTACHMENT NO. 1 TO TYPE B VARIABLES TABLE ITEM NO. B-14

CONTAINMENT ISOLATION VALVES

Plant ID	Channel
VCR-20	Glycon Return from Containment
VCR-21	Glycon Return from Containment
XCR-100	Cntrl Air to Cntnment Islation Vlvs
XCR-101	Cntrl Air to Cntnment Islation Vlvs
XCR-102	Cntrl Air to Cntnment Islation Vlvs
XCR-103	Cntrl Air to Cntnment Islation Vlvs
GCR-301	N2 Supply to Pressurizer Relief Tnk
CCM-451	CCW fr RCPs Lwr Guide Bearing Coolr
CCM-452	CCW fr RCPs Lwr Guide Bearing Coolr
CCM-453	CCW from RCPs Thermal Barriers
CCM-454	CCW from RCPs Thermal Barriers
CCM-458	CCW to RCPs Oil Coolers and Thermal
CCM-459	CCW to RCPs Oil Coolers and Thermal Barriers
ECR-31	Containment Air Monitor
ECR-32	Containment Air Monitor
ECR-33	Containment Air Monitor
ECR-35	Containment Air Monitor
ECR-36	Containment Air Monitor
WCR-901	NESW to CLV Unit 1
WCR-905	NESW to CLV Unit 2
WCR-909	NESW to CLV Unit 3
WCR-913	NESW to CLV Unit 4
WCR-900	NESW to RCP CLV Unit 1
WCR-904	NESW to RCP CLV Unit 2
WCR-908	NESW to RCP CLV Unit 3
WCR-912	NESW to RCP CLV Unit 4
WCR-902	NESW from CLV Unit 1
WCR-903	NESW from CLV Unit 1
WCR-906	NESW from CLV Unit 2
WCR-907	NESW from CLV Unit 2
WCR-910	NESW from CLV Unit 3
WCR-911	NESW from CLV Unit 3
WCR-914	NESW from CLV Unit 4
WCR-915	NESW from CLV Unit 4
WCR-921	NESW to CUV Unit 1
WCR-925	NESW to CUV Unit 2
WCR-929	NESW to CUV Unit 3
WCR-933	NESW to CUV Unit 4
WCR-920	NESW to RCP CUV Unit 1
WCR-924	NESW to RCP CUV Unit 2
WCR-928	NESW to RCP CUV Unit 3
WCR-932	NESW to RCP CUV Unit 4
WCR-922	NESW from CUV Unit 1
WCR-923	NESW from CUV Unit 1
WCR-926	NESW from CUV Unit 2
WCR-927	NESW from CUV Unit 2
WCR-930	NESW from CUV Unit 3
WCR-931	NESW from CUV Unit 3
WCR-934	NESW from CUV Unit 4
WCR-935	NESW from CUV Unit 4
WCR-941	NESW to RCP 1 Air Cooler
WCR-942	NESW to RCP 2 Air Cooler
WCR-943	NESW to RCP 3 Air Cooler
WCR-944	NESW to RCP 4 Air Cooler
WCR-945	NESW from RCP 1 Air Cooler
WCR-955	NESW from RCP 1 Air Cooler
WCR-946	NESW from RCP 2 Air Cooler
WCR-956	NESW from RCP 2 Air Cooler
WCR-947	NESW from RCP 3 Air Cooler
WCR-957	NESW from RCP 3 Air Cooler
WCR-948	NESW from RCP 4 Air Cooler
WCR-958	NESW from RCP 4 Air Cooler
WCR-951	NESW to RCP 1 Air Cooler
WCR-952	NESW to RCP 2 Air Cooler
WCR-953	NESW to RCP 3 Air Cooler
WCR-954	NESW to RCP 4 Air Cooler
WCR-960	NESW to Instrmnt Rm Vntilatn Units
WCR-961	NESW to Instrmnt Rm Vntilatn Units
WCR-964	NESW to Instrmnt Rm Vntilatn Units
WCR-965	NESW to Instrmnt Rm Vntilatn Units
WCR-962	NESW fr Instrmnt Rm Vntilatn Units
WCR-963	NESW to Instrmnt Rm Vntilatn Units
WCR-966	NESW to Instrmnt Rm Vntilatn Units
WCR-967	NESW fr Instrmnt Rm Vntilatn Units
CCR-440	CCW from Main Steam Penetration
CCR-441	CCW from Main Steam Penetration
MCM-221	Main Steam to Auxiliary Feed Pump
MCM-231	Main Steam to Auxiliary Feed Pump
CCM-430	CCW to E Pressure Equalization Fan
CCM-431	CCW to E Pressure Equalization Fan
CCM-432	CCW to W Pressure Equalization Fan
CCM-433	CCW to W Pressure Equalization Fan

ATTACHMENT NO. 1 TO TYPE B VARIABLES TABLE ITEM NO. B-14

CONTAINMENT ISOLATION VALVES

Plant ID	Channel
CCR-455	CCW to Reactor Supports
CCR-456	CCW from Reactor Supports
CCR-457	CCW from Reactor Supports
CCR-460	CCW from Excess Letdown Hx
CCR-462	CCW to Excess Letdown Hx
DCR-201	RC Drain Tank to Vent Header
DCR-202	RC Drain Tank to Gas Analyzer
DCR-203	RC Drain Tank to Vent Header
DCR-204	RC Drain Tank to Gas Analyzer
DCR-205	RC Drain Tank Suction Isolation
DCR-206	RC Drain Tank Suction Isolation
DCR-207	Nitrogen Supply to RC Drain Tank
DCR-301	S/G Blowdown Sample Isolation
DCR-302	S/G Blowdown Sample Isolation
DCR-303	S/G Blowdown Sample Isolation
DCR-304	S/G Blowdown Sample Isolation
DCR-310	S/G #1, #2, #3 & #4 Blowdown Valves
DCR-320	S/G #1, #2, #3 & #4 Blowdown Valves
DCR-330	S/G #1, #2, #3 & #4 Blowdown Valves
DCR-340	S/G #1, #2, #3 & #4 Blowdown Valves
DCR-600	Containment Sump to Waste Holdup
DCR-601	Containment Sump to Waste Holdup
DCR-610	Ice Condenser Drain to Drain Header
DCR-611	Ice Condenser Drain to Drain Header
DCR-620	Cntnment Ventilation Drain to Hldup
DCR-621	Cntnment Ventilation Drain to Hldup
ECR-416	Containment Liquid Sampling System
ECR-417	Containment Liquid Sampling System
ECR-496	Containment Liquid Sampling System
ECR-497	Containment Liquid Sampling System
ECR-535	Containment Gas Sampling System
ECR-536	Containment Gas Sampling System
QCR-919	Cont. Demin. Cleanup Water Isol.
QCR-920	Cont. Demin. Cleanup Water Isol.

Plant ID	Channel
ECR-10	Cntnment H2 Sample & Return Valves
ECR-11	Cntnment H2 Sample & Return Valves
ECR-12	Cntnment H2 Sample & Return Valves
ECR-13	Cntnment H2 Sample & Return Valves
ECR-14	Cntnment H2 Sample & Return Valves
ECR-15	Cntnment H2 Sample & Return Valves
ECR-16	Cntnment H2 Sample & Return Valves
ECR-17	Cntnment H2 Sample & Return Valves
ECR-18	Cntnment H2 Sample & Return Valves
ECR-19	Cntnment H2 Sample & Return Valves
ECR-20	Cntnment H2 Sample & Return Valves
ECR-21	Cntnment H2 Sample & Return Valves
ECR-22	Cntnment H2 Sample & Return Valves
ECR-24	Cntnment H2 Sample & Return Valves
ECR-23	Cntnment H2 Sample & Return Valves
ECR-25	Cntnment H2 Sample & Return Valves
ECR-26	Cntnment H2 Sample & Return Valves
ECR-27	Cntnment H2 Sample & Return Valves
ECR-28	Cntnment H2 Sample & Return Valves
ECR-29	Cntnment H2 Sample & Return Valves
GCR-314	Nitrogen Supply to Accumulators
ICR-5	Accumulator Sample Valves
ICR-6	Accumulator Sample Valves
MCR-251	Sample from Main Steam Lines
MCR-252	Sample from Main Steam Lines
MCR-253	Sample from Main Steam Lines
MCR-254	Sample from Main Steam Lines
NCR-105	Primary System Hot Leg Sample
NCR-106	Primary System Hot Leg Sample
NCR-107	Pressurizer Liquid Sample
NCR-108	Pressurizer Liquid Sample
NCR-109	Pressurizer Steam Sample
NCR-110	Pressurizer Steam Sample
QCR-300	Letdown Line Isolation Valve
QCR-301	Letdown Line Isolation Valve
NCR-252	Prmary Mkup H20 to Prssrzer Rlf Tnk
QCM-250	RCP Seal H20 Return Isolation Valve
QCM-350	RCP Seal H20 Return Isolation Valve
RCR-100	Prssrzer Rlf Tnk to Gas Analyzer
RCR-101	Prssrzer Rlf Tnk to Gas Analyzer
VCR-10	Glycon Supply to Containment
VCR-11	Glycon Supply to Containment

ATTACHMENT NO. 2 TO TYPE B VARIABLES TABLE

ITEM NO. B-14

Most of the valves listed in Attachment No. 1 to Type B Variables Item No. B-14 are located outside of containment and are only subject to an HELB outside of containment. Since they are not required to operate (per EOP's) in the event of an HELB, qualification is not required because they will be located in a mild environment should a Design Basis Event (DBE) occur where their use is required. Redundant indication is also not required because the valves are backed up by a second, redundant valve, also listed in Attachment No. 1.

Exceptions to the above are listed as follows:

DCR-301,302,303,304,310,320,330,340, and XCR-100,101,102,103 will be upgraded by replacing the position indication limit switches with environmentally and seismically qualified equipment. These valves are located outside of containment and are required to operate during an HELB.

The position indication limit switch for QCM-250 is located outside containment and is not qualified for a DBE inside of containment. It is, however, backed up by QCM-350 which is located outside of containment and its position indication limit switch is qualified. Should there be a DBE inside containment and QCM-250 did not indicate appropriately, QCM-350 could be used to verify isolation. In addition an operator could be dispatched to visually verify QCM-350 closure. If a DBE outside of containment should occur, QCM-350 is qualified for the adverse environment generated and could be backed up by QCM-250 which would not be subjected to the harsh environment and therefore, would be expected to indicate appropriately. See footnote (ee)

The position indication limit switches for VCR-11 and 21 are located inside containment and not qualified for a DBE inside containment. They are, however, backed up by VCR-10 and 20, respectively, which are located outside of containment. Should there be a DBE inside containment and VCR-11 or 21 did not indicate appropriately, VCR-10 and 20 can be used to verify isolation. In addition, an operator could be dispatched to visually verify closure. DBE's outside of containment will not create a harsh environment at VCR-10 and 20 locations. See footnote (ee)

CCM-451,452,453,454,458,459; CCM-430,431,432,433; and MCM-221,231 are qualified for an HELB.

VCR-101 thru 107 and 201 thru 207 are not listed in Attachment 1 because their function is specifically listed in the Type D Variables Table, Item D-31.

4.0 FOOTNOTES DESIGNATING DEVIATIONS TO REGULATORY GUIDE 1.97, REV. 3

(a) The automatic injection of boric acid into the RCS by the safety injection system following a postulated LOCA/HELB will be monitored and verified through the use of qualified instrumentation. In addition, since all sources of water for the safety injection system (Accumulators, Boron Injection Tank and Refueling Water Storage Tanks) are required by Technical Specifications to contain boric acid solution of a minimum concentration, the proper operation of the ECCS ensures an adequate boron concentration in the reactor coolant to achieve and maintain the safe shutdown of the reactor core. The RCS soluble boron content is not expected to change rapidly, if at all, following the initial borating during the SI phase of an accident. Periodic analysis of RCS samples would detect any significant changes in boron concentration. Instrumentation to continuously monitor RCS soluble boron concentration is not required since periodic analysis of RCS grab samples is adequate for verification of reactivity control. This is a deviation from the recommendations of Reg. Guide 1.97 Rev. 3 for providing continuous RCS boron concentration indication (Deviation No. DV-8 - Also see footnote (bb)).

(b) Redundancy not required per NUREG 0737 requirements.

(c) Seismic qualification not required per NUREG 0737 requirements.

(d) All equipment when required to be used, is located in a mild environment, therefore environmental qualification is not required.

(e) Lack of lower containment spray flow monitoring instrumentation will not deter the operator's ability to determine adverse containment conditions. adverse containment conditions can be monitored by looking at items such as containment pressure. If we see containment pressure conditions different than what is expected, then we can confirm whether adverse containment conditions are due to a lack of spray flow by the monitoring of containment spray pump discharge pressure.

(f) DELETED

(g) DELETED

(h) Instrumentation to continuously monitor RCS radioactivity is not required. See footnote (ff). Periodic analysis of RCS grab samples is adequate to detect deterioration of fuel cladding. Indicative of an inadequate core cooling (ICC) event, fast deterioration of fuel cladding could be detected by sensing the ICC conditions through diverse instrumentation (i.e., RVLIS, CET's, TSAT meter).

(i) Per SER issued June 16, 1981 concerning Auxiliary Feedwater System reliability, it is only required that qualified, S/G narrow range level indication be provided and backed up by qualified Auxiliary Feedwater Flow Indication. S/G wide range level indication is not required to be environmentally qualified or powered from a Class IE power source.

(j) DELETED

(k) DELETED

(l) INTENTIONALLY LEFT BLANK

(m) DELETED

(n) DELETED

(o) INTENTIONALLY LEFT BLANK

(p) DELETED

(q) Not required per NUREG 0737 II.B.3

(r) These valves are left normally open (safe position), the breakers racked out and they cannot change position. Therefore, Environmental Qualification of position indication is not required.

(s) Redundancy can be provided by VCR-101 thru 107 which are located inside containment. VCR-201 thru 207 are located outside containment.

(t) Our original submittal identified both the centrifugal charging pump (CCP) flow and CCP motor breaker status as type A variables. This would require Category 1 instrumentation for monitoring these parameters. The CCP breaker status indication will meet the Regulatory Guide 1.97, Rev. 3 recommendations for Category 1 instrumentation, except as noted in the Table under item No. A.28.

With regard to the CCP flow indication, it should be noted that our Emergency Operating Procedures require manual operator action based on indication of pump operation or flow. The CCP breaker status indication and other parameters serve to verify pump operation. The non Category 1 CCP flow indication can serve as a backup.

The above is a basis for our deviation from the Regulatory Guide 1.97, Rev 3 recommendation to provide Category 1 instrumentation for CCP flow indication. (This deviation was discussed in AEP:NRC:0773S, Attachment 3, Item 2).

(u) Transmitters to be replaced to improve accuracy. See AEP:NRC:0836T dated July 16, 1987.

(v) Credit was not originally taken for the seismic qualification of equipment as noted in AEP:NRC:0773O. We will take credit for seismic qualification of this equipment which has been established. This response was also given in AEP:NRC:0773S Attachment 1, Item 3.3.7.

(w) Monitoring of containment pressure is currently provided by two Category 3 wide-range (-5 to 36 psig instruments) and four Category 3 narrow-range (-5 to 12 psig) instruments. The design pressure of the Cook Nuclear Plant containments is 12 psig. The four narrow-range instruments are scheduled to be upgraded to meet Category 1 requirements by the end of the 1987 refueling outage for Unit 1 (which has been completed) and by the end of the 1988 refueling outage for Unit 2.

The wide-range containment pressure instrumentation ranges were revised to meet the requirements of NUREG-0578 and NUREG-0737. These instruments are not powered by a emergency standby power source as recommended by Regulatory Guide 1.97, Rev. 3 for Category 1 instrumentation, and they do not meet the Category 1 separation criteria. The wide-range instrumentation is, however, highly reliable, and as a result we believe it is unlikely that it would not be available if needed to monitor the course of an accident. Further, it is our belief that for other than short-term individual compartment pressure peaks, the narrow-range instrumentation would span the range of pressure anticipated in our evaluation of loss-of-coolant-type accidents. On the above basis, this is a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 wide-range containment pressure instrumentation (Deviation No. DV-6). See response to AEP:NRC:0773S Attachment 3, Item 4.

(x) The saturation meter equipment was originally installed in accordance with the requirements of NUREG-0578. In an SER dated March 20, 1980, the equipment installed to monitor degrees of subcooling was found to be acceptable (NRC letter, A. Schwencer to John E. Dolan, dated March 20, 1980). As noted in that correspondence, the device installed was a discrete digital monitor, and the plant process computer was used in conjunction with this monitor to provide subcooling margin. Additionally, as part of the NUREG-0737 Supplement 1 requirements, a subcooling margin curve is provided by our Technical Support Center computer. We believe that these three instrument systems, which have been installed to be consistent with the requirements of their appropriate documents, are reliable. As a result, we believe it is unlikely that they would not be available if needed to monitor the course of an accident.

It should be noted that our original submittal providing status of Regulatory Guide 1.97, Rev. 3 compliance (AEP:NRC:07730, dated October 15, 1985) was intended to identify instrumentation currently in compliance with the Regulatory Guide; instrumentation not in compliance for which upgrading to the Regulatory Guide recommendations was planned; and instrumentation not in compliance for which justification for a deviation from the Regulatory Guide recommendations was provided. The instrumentation for monitoring degrees of subcooling falls into the latter category.

Based on the above information and justification, we again submit a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 instrumentation for monitoring degrees of subcooling (Deviation No. DV-7). See AEP:NRC:0773S Attachment 1, Item 3.3.3.

(y) For similar reasons to those stated in (t) above, we submit a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide Category 1 instrumentation for safety injection (SI) pump flow indication (Deviation No. DV-22). The SI pump motor breaker status instrumentation will meet Category 1 requirements except as noted in the Table under Item A.29. This deviation was detailed in AEP:NRC:0773S Attachment 3, Item 3.

(z) Our original submittal identified the steam generator blow down radiation indication as a Type A variable. However, because of changes in our Emergency Operating Procedures made subsequent to our original submittal,

manual operator action is no longer based on this variable. We therefore request that steam generator blowdown radiation indication be deleted from our original list of Type A variables. This response was given in AEP:NRC:0773S Attachment 3, Item 6.

(aa) We will provide Neutron Flux Monitoring to comply with Reg Guide 1.97 Rev 3 recommendations, except as noted in the Table.

(bb) Primary coolant boron concentration can be measured in a range of 375 ppm to 10,000 ppm. This range is based on PASS reactor coolant samples with a 1:1000 dilution. The undiluted reactor coolant grab sample will be measured in the .375 to 10 ppm range.

PASS would be used during and following loss-of-coolant accidents. In the event of a LOCA, emergency boration and injection from the refueling water storage tank would occur and we would therefore expect a reactor coolant boron concentration substantially in excess of the low range of our PASS sample measurement capability.

On the basis of the above, we submit a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to provide the capability to measure boron concentration in PASS samples to the lower limit of 0 ppm (Deviation No. 8). This deviation was also detailed in AEP:NRC:0773S Attachment 1, Item 3.3.2.

(cc) Our original submittal indicated that we would replace the cold and hot leg RCS water temperature recorders with Category 1 instruments by the end of the 1987 refueling outages for Units 1 and 2. However, a more detailed review of our current design has resulted in the identification of additional work (e.g., control room equipment and cable relocation, and installation of new emergency standby power sources) that needs to be performed beyond that identified at the time of our original submittal. We therefore request that the completion dates for upgrading the recorders to meet Category 1 requirements be changed to the 1989 refueling outage for Unit 1 and 1990 refueling outage for Unit 2. This was noted in AEP:NRC:0773S Attachment 3, Item 5. We have also since determined that only two hot leg and two cold leg channels are necessary to comply with the Reg. Guide 1.97 recommendations and therefore we are only taking credit for the instruments noted in the table. This item is pending NRC approval of an Appendix R change request.

(dd) Because of changes to the EOP's from the original October, 1985 submittal and planned changes to this instrumentation, we have reclassified this instrumentation as Category 2 per Reg. Guide 1.97 guidance.

(ee) We will upgrade the valve position limit switches on valves VCR-11 and VCR-21 to meet the environmental qualification requirements of 10 CFR 50.49 and Regulatory Guide 1.97, Rev. 3 recommendations.

The valve position limit switches for valve QCM-250, as well as the associated cable and terminations, are qualified in accordance with 10 CFR 50.49(k) except that they have not been qualified for submergence. The QCM-250 position indication limit switch is located below maximum flood level. Although this is not completely consistent with the Regulatory Guide 1.97, Rev. 3 recommendations for equipment qualification, we do not believe any upgrading of the position indication limit switch is necessary.

This is due to the fact that QCM-250 is designed to close within 15 seconds of a containment isolation signal, which means that the valve will not become submerged before it performs its safety function. In addition, once the valve is closed, it is extremely unlikely that it would change position due to its submergence.

Given these considerations, we believe that QCM-250 in its present status, without upgrading, adequately meets the intent of Regulatory Guide 1.97, Rev. 3 recommendations for achieving verifiable containment isolation. We therefore submit a deviation from the recommendation of Reg. Guide 1.97 Rev. 3 with respect to environment qualification for QCM-250. (Deviation No. DV-24).

The planned schedule for upgrading VCR-11 and VCR-21 to meet 10 CFR 50.49 requirements calls for this work to be completed in both units by the end of the refueling outages presently scheduled for 1989 (Unit 1) and 1990 (Unit 2).

This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.4.

(ff) As stated in our original submittal (AEP:NRC:07730, dated October 15, 1985) the primary coolant system radioactivity is not continuously monitored by in-line instrumentation. Rather, periodic analysis of reactor coolant grab samples is provided to detect deterioration of fuel cladding. Our post-accident sampling system provides a diluted grab sample which is analyzed by the gamma spectrum analyzer. See our response to footnote (gg) below for the range of our gamma spectrum analyzer. On the basis of the sampling capability described in footnote (gg) we are submitting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for continuous monitoring of radioactivity in the reactor coolant system (Deviation No. DV-9). It should also be noted that Category 1 requirements for this system are only to be applicable to equipment that operates equipment installed in the portion of piping that is Seismic Class I. Electrical equipment operating equipment installed in Seismic Class 3 piping is to meet Category 3 requirements. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.5.

(gg) We believe the EG&G evaluation should have cited a range of 1 uCi/ml to 10 Ci/ml for this variable as per Table 3 of Regulatory Guide 1.97, Rev. 3. Our range of measurement for gamma spectrum analysis of the diluted post-accident system grab samples of primary coolant is 1 uCi/ml to 10 Ci/ml. This complies with the Regulatory Guide 1.97, Rev. 3 recommended range.

(hh) The instrumentation identified for this parameter in our original submittal was incorrect. We initially identified our lower containment normal process radiation monitors (ERS-1300 and 1400 for Unit 1 and ERS-2300 and 2400 for Unit 2) for this parameter. The instruments used to monitor this parameter are the unit vent radiation monitors (VRS-1500 for Unit 1 and VRS-2500 for Unit 2). The display location for these instruments is the control room CT-1 control terminal, not panel WDG. All other information contained in our original submittal for this item remains the same. This information was provided in AEP:NRC:0773S Attachment 3, Item 7.

(ii) We incorrectly identified the instruments as ITR-311 and 321. These are actually the RHR heat exchanger inlet temperature devices. The outlet instrumentation tag numbers are ITI-310 and 320. These devices have a range of 0-400°F and indicate locally and in the Technical Support Center computer terminal located in the Technical Support Center and the control room. We believe this instrumentation meets the Regulatory Guide 1.97 recommendations for RHR heat exchanger outlet temperature measurement range. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.10.

(jj) The correct range for IFI-310 and 320 is 1500-5000 GPM. We inadvertently reported the incorrect range to you in our previous submittal AEP:NRC:-0773O. We believe that this still complies with Reg. Guide 1.97 Rev. 3 requirements for range for this variable.

(kk) To clarify our initial response, these motor-operated valves are normally left in the open position when the plant is operating in Mode 1 and Mode 2. The circuit breakers are racked out and the valves cannot, therefore, spuriously change position. They can change position only as the result of deliberate operator action. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.12.

(ll) We presently have instrumentation installed that we believe meets Regulatory Guide 1.97, Rev. 3 recommendations. Pressurizer heater current can be monitored by observing ammeters located on the pressurizer control panel in the control room. The range is 0-200 amps. The pressurizer heaters are powered from the safety buses and therefore have the capability of being powered by the emergency power sources. Automatic shedding of the pressurizer heaters following a blackout is provided to prevent overloading of the emergency power sources. The operator can afterwards, at his discretion, manually energize the pressurizer heaters, taking care not to overload the emergency power sources. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.14.

(mm) We do not rely on the quench tank to perform any post-pressurizer release function. However, we are providing the following information in response to the EG&G evaluation. The range of 74% of total tank volume originally submitted was not accurately stated to show the adequacy of the existing installation. The correct range should have been stated as being from 7 inches above the tank bottom to 7 inches below the tank top. This range includes coverage of the sparger. With regard to the ability to quench a "design-basis" pressurizer release, as noted above we do not rely on the quench tank to perform this function. The quench tank is used during normal plant operation to contain pressurizer releases from routine pressurizer pressure adjustments and valve leakage. In the case of a design-basis event that causes the PORVs and safety relief valves to lift, two rupture disks will burst before reaching the quench tank design pressure of 100 psi. Subsequently discharge through the quench tank into the containment sump will occur.

With regard to overpressurization, we do not understand the basis for the EG&G position that sufficient gas volume exists to accept pressurizer release without becoming overpressurized. As noted above, overpressurization will not occur, because rupture discs will burst and discharge into the containment before reaching the tank design pressure of 100 psig.

Normal water level is kept at between 80% and 84% of the instrument range with a high alarm at 84% and a low alarm at 79%. As such, in-leakage from the relief discharge system can be adequately monitored. We therefore submit a deviation from the Regulatory Guide 1.97 recommendation to monitor quench tank level from top to bottom of tank (Deviation No. DV-10). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.15.

(oo) On August 21, 1981 we submitted a letter (AEP:NRC:0300G) that documented discussions with NRR staff clarifying certain portions of an NRC SER (June 16, 1981) of the Cook Nuclear Plant auxiliary feedwater system. In that letter it was confirmed that Regulatory Guide 1.97 recommendations for steam generator level instrumentation did not have to be implemented at that time, but that implementation would be addressed at some time in the future through the Regulatory Guide 1.97 compliance/commitment process.

The steam generator wide-range level indication is not required for post-accident monitoring and in fact has been deleted from our Technical Specifications on Units 1 and 2. This was stated in our December 10, 1980 letter, which submitted a proposed amendment to our Technical Specifications (AEP:NRC:0449). As stated in that letter, the reasons for deletion of steam generator wide-range level indication from the Technical Specifications are: (1) the S/G wide-range level indication does not perform any safety-related function and is not assumed operable in the various plant safety analyses; and (2) the S/G narrow-range instrumentation, which we believe fulfills post-accident monitoring requirements, is environmentally and seismically qualified, powered from a Class 1E source and has three redundant channels per S/G. The S/G level indication is backed up by auxiliary feedwater flow instrumentation.

The S/G wide-range level instrumentation is powered from a Class 1E source, but all four channels are powered by the same source. Since this is not in compliance with the Regulatory Guide 1.97, Rev. 3 recommendation, and based on the information given above, we are submitting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for S/G level instrumentation. (Deviation No. DV-11) This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.16.

(pp) We incorrectly identified the range of instrumentation measuring this variable as $0\text{-}5 \times 10^6$ PPH. It should read $0\text{-}4 \times 10^6$ PPH. We believe that this still complies with Reg. Guide 1.97 Rev. 3 requirements for range for this variable.

(qq) I&MECo currently provides condensate storage tank (CST) level indication in the control room through three highly reliable Category 3 level-measuring devices. One of these instruments is electrically operated, while the other two are pneumatic devices. In addition, CST level can be read at the local turbine-driven auxiliary feedwater pump control panel. I&MECo has also committed to provide additional CST level indication by adding an instrument channel to meet Category 1 requirements. The new channel was installed during the 1987 refueling outage on Unit 1 and the 1988 refueling outage on Unit 2.

Normal water level is kept at between 80% and 84% of the instrument range with a high alarm at 84% and a low alarm at 79%. As such, in-leakage from the relief discharge system can be adequately monitored. We therefore submit a deviation from the Regulatory Guide 1.97 recommendation to monitor quench tank level from top to bottom of tank (Deviation No. DV-10). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.15.

(oo) On August 21, 1981 we submitted a letter (AEP:NRC:0300G) that documented discussions with NRR staff clarifying certain portions of an NRC SER (June 16, 1981) of the Cook Nuclear Plant auxiliary feedwater system. In that letter it was confirmed that Regulatory Guide 1.97 recommendations for steam generator level instrumentation did not have to be implemented at that time, but that implementation would be addressed at some time in the future through the Regulatory Guide 1.97 compliance/commitment process.

The steam generator wide-range level indication is not required for post-accident monitoring and in fact has been deleted from our Technical Specifications on Units 1 and 2. This was stated in our December 10, 1980 letter, which submitted a proposed amendment to our Technical Specifications (AEP:NRC:0449). As stated in that letter, the reasons for deletion of steam generator wide-range level indication from the Technical Specifications are: (1) the S/G wide-range level indication does not perform any safety-related function and is not assumed operable in the various plant safety analyses; and (2) the S/G narrow-range instrumentation, which we believe fulfills post-accident monitoring requirements, is environmentally and seismically qualified, powered from a Class 1E source and has three redundant channels per S/G. The S/G level indication is backed up by auxiliary feedwater flow instrumentation.

The S/G wide-range level instrumentation is powered from a Class 1E source, but all four channels are powered by the same source. Since this is not in compliance with the Regulatory Guide 1.97, Rev. 3 recommendation, and based on the information given above, we are submitting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for S/G level instrumentation. (Deviation No. DV-11) This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.16.

(pp) We incorrectly identified the range of instrumentation measuring this variable as $0\text{-}5 \times 10^6$ PPH. It should read $0\text{-}4 \times 10^6$ PPH. We believe that this still complies with Reg. Guide 1.97 Rev. 3 requirements for range for this variable.

(qq) I&M currently provides condensate storage tank (CST) level indication in the control room through three highly reliable Category 3 level-measuring devices. One of these instruments is electrically operated, while the other two are pneumatic devices. In addition, CST level can be read at the local turbine-driven auxiliary feedwater pump control panel. I&M has also committed to provide additional CST level indication by adding an instrument channel to meet Category 1 requirements. The new channel was installed during the 1987 refueling outage on Unit 1 and the 1988 refueling outage on Unit 2.

The CST is the initial source of water for the auxiliary feedwater (AFW) system, and provides sufficient volume to maintain the reactor coolant system in a hot standby condition for 9 hours. In the event that sufficient water is not available from the CST in one unit, operating procedures call for a cross-tie valve to be opened to supply feedwater from the CST in the other unit.

In the unlikely event that neither CST can supply sufficient AFW, procedures require transferring the supply source to the essential service water system (ESWS). The water supply for the ESWS is Lake Michigan.

In view of the number and diversity of instrumentation available to provide CST level monitoring, and the ultimate availability of Lake Michigan as a source of auxiliary feedwater, we are submitting a deviation from the Regulatory Guide 1.97 recommendation to provide more than one Category 1 level indication for the CST (Deviation No. DV-12). This information was provided in AEP:NRC:0773S Attachment 3, Item 1.

(rr) When operating normally, each containment spray pump will deliver 3200 gpm (design flow) at 490 ft. TDH. We have attached to this response a curve showing containment spray pump flow as a function of pump discharge pressure (drawing No. HXP87055JW-1). See Attachment No. 4.

The attached curve indicates the expected range of operation for the containment spray pumps. This operating range stems from consideration of pump suction head, containment pressure, and pump operating characteristics. Routine surveillance of spray pump operation is performed to ensure that, if containment spray is required, the pumps will operate in the indicated area of the flow curve and hence provide the necessary flow to the containment spray system. The reactor operators can, therefore, verify proper containment spray flow by monitoring spray pump discharge pressure to confirm that it is within the expected range.

It should be noted that the upper containment spray flow instrumentation cited in our original submittal (AEP:NRC:0773O, dated October 15, 1985 [IFI-330 and 331]) measures only the flow provided by the RHR pumps to the upper containment spray, not the flow from the containment spray pumps. However, the containment spray pumps, not the RHR pumps, are normally used to supply containment spray flow. Also, please note that the flow range of 0-200 gpm for IFI-330 and 331 (for measurement of RHR pump flow to the upper containment spray) given in that submittal is incorrect. The correct range is 0-2500 gpm.

Based on the above, we are noting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for containment spray flow instrumentation (Deviation No. DV-13). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.17.

(ss) The RHR heat exchanger inlet temperature instrumentation will be upgraded to meet Regulatory Guide 1.97 Rev. 3 recommendations except as noted in the tables. This response was given in AEP:NRC:0773S Attachment 1, Item 3.3.18.

(tt) Because of the following actions which apply for normal, accident, and post-accident conditions, we believe level indication beyond that currently provided is not required. Upon receiving a hi-level alarm, flow into the Volume Control Tank (VCT) is automatically fully diverted into the hold-up tanks. If a low-level alarm is reached, an alarm alerts the operator to restore level. In the event this effort fails, an emergency lo-lo level alarm is sounded and the refueling water sequence is automatically initiated. We believe that this range (0-70 inches) is adequate to safely monitor the operation of this tank. In the unlikely event that VCT level indication is lost and the VCT becomes completely full, a safety relief valve (set at 75 psig) will open and the excess water will be discharged into the hold-up tanks. We therefore are submitting a deviation from the Regulatory Guide 1.97 recommendations to monitor Volume Control Tank Level from top to bottom (Deviation No. DV-14). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.19.

(uu) An analysis previously performed for one of the radiation exposure rate monitors (VRC-301) listed in our original submittal (AEP:NRC:07720, dated October 15, 1985) showed that the exposure rate range of 10^{-1} mR/hr to 10^{4} mR/hr was adequate to monitor plant operation in the area in which this monitor is installed.

As part of a general upgrade of area radiation monitors at the Cook Nuclear Plant, monitor numbers NRA-340 and RRA-332 will be replaced. As previously noted in our submittal AEP:NRC:0773S, Item 3.3.9, concurrent with the upgrade activities, analyses of the type mentioned above was to be performed to determine what range of exposure rate measurement is appropriate for the monitors in the areas where they are to be installed. We expected that the analysis would show that a range less than that recommended by Regulatory Guide 1.97, Rev. 3 would be adequate to safely monitor plant operations in the areas where these monitors are to be installed. This analysis has been completed and the results are explained in footnote (kkk) herein. The upgrading program is scheduled for completion at both Unit 1 and Unit 2 by the end of 1989.

(vv) INTENTIONALLY LEFT BLANK

(ww) The range for SFR-401 was incorrectly identified as 0-2000 SCFM. The correct range for measurement of this variable is 0-250 SCFM. We believe that this still complies with the Reg. Guide 1.97 Rev. 3 requirements for range for this variable.

(xx) This instrumentation was recently (1985) upgraded by the addition of a high-range noble gas detector. Based on our recent primary calibration analysis, the range of this instrumentation was determined to be 5.8×10^{-7} uCi/cc to 1.86×10^{4} uCi/cc Xenon-133 dose equivalent. On July 23, 1986 a letter was sent to the NRC (AEP:NRC:0678Y) in which we stated that post-accident conditions would not result in steam jet air ejector exhaust noble gas concentration greater than 2×10^{3} uCi/cc. On this basis we requested an exemption from the NUREG-0737, Section II.F.1-1 upper-range requirement of 10^{5} uCi/cc in favor of a more realistic upper range of 10^{4} uCi/cc. We therefore are submitting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation for this parameter (Deviation No. DV-15).

It should also be noted that the tag numbers SFR-1900 and SFR-2900 given in our original submittal are incorrect. The correct tag numbers are SRA-1900 and SRA-2900. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.21.

(yy) The range of 3 uCi/cc to 20 x 10^5 uCi/cc as provided in our submittal was based solely on the monitor's response to Xe-133 and not to the anticipated mixture of radioisotopes following a steam generator tube rupture. The lower limit of 0.1 uCi/cc of Xe-133 equivalent mixture can be measured. As stated in our September 8, 1986 letter (AEP:NRC:0678Z), when the anticipated mixture of radioisotopes for a steam generator tube rupture is used, the maximum concentration is calculated to be 0.263 uCi/cc Xe-133 equivalent activity. With respect to this upper range limit, an exemption from the NUREG-0737 requirement of 1000 uCi/cc was requested in the September 8, 1986 letter and a 100 uCi/cc value proposed. No response to our request has been received at this writing. We are submitting the same upper limit deviation from the Regulatory Guide 1.97, Rev. 3 guidelines (Deviation No. DV-16). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.22.

(zz) It was planned to have available by the end of 1988 a portable gamma-ray spectroscopy system providing the capability for field analysis of plant and environs radioactivity. This equipment was shipped to the Donald C. Cook Nuclear Plant on July 30, 1987 and is now available for use. Information related to this variable was provided in AEP:NRC:0773S Attachment 1, Item 3.3.23.

(aaa) The capability to measure gross activity in the range of 1 uCi/ml - 10 Ci/ml is available; however, this measurement is not normally used to assess core damage. Rather, our initial core damage assessment is done through gamma spectrum analysis of primary coolant. This method provides an isotopic analysis as well as giving an indication of total primary coolant activity. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.24 No. 1.

(bbb) Gamma spectrum isotopic analysis is performed in an energy range of 0.050-2.05 MeV. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.24 No. 2.

(ccc) Boron content is measured in the range of 375-10,000 ppm (see footnote (bb)) This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.24 No. 3.

(ddd) Chloride content in undiluted samples 30 days after an accident is measured in a range of 0.01 to 20 ppm. For diluted samples 1:1000 taken within 4 days of an accident, the range of measurement is 10 to 20,000 ppm.

We are noting a deviation from the Regulatory Guide 1.97, Rev. 3 lower limit of 0 ppm (Deviation No. DV-17). This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.24 No. 4.

(eee) Our range of measurement of pH is 1 to 13. Our original submittal showed a range of 5 to 8 which we believed to be the range of interest for this parameter. This information was provided in AEP:NRC:0773S Attachment 1, No. 5.

(fff) An exemption from the requirement for taking hydrogen grab samples of containment air was granted via letter from Youngblood (NRC) to Dolan (AEP) dated November 5, 1986. Therefore, we are also submitting a similar deviation with respect to Reg. Guide 1.97 Rev. 3 (Deviation No. DV-18). We do, however, perform continuous monitoring of containment air hydrogen content in the range of 0 to 30 volume percent (see Item C-10) This information was provided in AEP:NRC:0773S Item 3.3.24 No. 6.

(ggg) NUREG-0737 does not require sampling of containment air oxygen content. As noted above, however, we do continuously monitor hydrogen content which mades containment air oxygen content of less concern from the standpoint of potential hydrogen flamability or deflagration. We therefore are submitting a deviation from the Regulatory Guide 1.97, Rev. 3 recommendation to sample for containment air oxygen content (Deviation No. DV-19). This information was provided in AEP:NRC:0773S Item 3.3.24 No. 7.

(hhh) We do not understand the EG&G request to provide containment air gamma spectrum "capacity." Regulatory Guide 1.97, Rev. 3 recommends that the capability be provided to perform an isotopic analysis of containment air. As recommended by Regulatory Guide 1.97, Rev. 3, gamma spectroscopy techniques are used to provide an isotopic analysis of noble gases in containment air. This gamma spectrum isotopic analysis is performed in a energy range of 0.050 to 2.05 MeV using a Canberra series 85 multichannel analyzer with a Digital PDP 11/24 computer and either a germanium (lithium-drifted) (Ge[Li]) or high-purity germanium (HPGe) detector. This information was provided in AEP:NRC:0773S Attachment 1, Item 3.3.24 No. 8.

(iii) Actual measured range was noted incorrectly in our previous submittal. Per our response in AEP:NRC:0773S Attachment 1, Item 3.3.11 we have made a minor correction to the actual measured range value. We believe that we still comply with the requirements for range for this variable.

(jjj) Circuits which require the use of indicating lamps for position indications, status indication, etc. will be using existing General Electric ET16 Indicating Lamps for this function. We have been advised by the manufacturer, that these indicating lamps meet their (the manufacturer's) interpretation of IEEE-344-1975. This indicating lamp is a seismically rugged commercial grade device for which comprehensive qualification is not available. Since these lamps are purchased as standard commercial grade material and are not manufactured for a specific order, 10CFR21 can not be applied to these devices. Therefore, we are submitting a deviation from Reg. Guide 1.97 for this device so we may continue to use and purchase it and its parts for use in monitoring Post Accident conditions (Deviation No. DV-20).

(kkk) ERA-7303 thru 7308, and ERA-8303 thru 8308 have ranges of 0.01 to 1000 R/HR. ERS-7401, 7403, 7404, 8401, ERA-7507, 7601, 7603, 7605 have ranges of .0001 to 10 R/HR. ERA-8403, 7504, 7508, 7602, 7604 have ranges of 0.001 to 10 R/HR. These are different than the recommended range of 10^{-1} to 10^{4} R/HR and we therefore are submitting a deviation for this variable in regards to range (Deviation No. DV-23). The justification for this request is as follows. With the exception of ERA-7305, 7306, 8305, and 8306, the worst case maximum estimated accident dose rate is less than the upper range limit noted above. The lower upper range limit is used to provide more accurate, useful information and to help prevent false "low fail" alarms.

In the case of ERA-7305, 7306, 8305, 8306, the worst case maximum estimated accident dose rate is 1730 R/HR which exceeds the upper range limit of 1000 R/HR. However, within one (1) hour, this drops to 573 R/HR which is well within the upper range limit. We believe that because personnel entry in an area where exposure may exceed 1000 R/HR (indicated by a "high fail" status indication) is highly unlikely and because the dose rate will quickly fall below the upper range limit of 1000 R/HR (at which time a quantitative indication will again be available) the range of 0.01 to 1000 R/HR is adequate. Again, using this range will provide more useful information and help prevent false "low fail" alarms.

(lll) If CCW water temperature is not available, adequate CCW cooling can be verified by monitoring CCW flow and RHR Inlet & Outlet temperatures, all of which are qualified (or planned to be qualified) for the intended purpose. Therefore, because of the availability of suitable diverse indications, environmental qualification of instrumentation monitoring this variable is not required. We are submitting a deviation with respect to Reg. Guide 1.97, Rev. 3 recommendations for this variable for environmental qualification, (Deviation No. DV-25).

(mmm) All equipment when required to be used, is located in a mild environment except for cables serving the following instruments. For these instruments the cables pass through harsh environment areas. Equipment qualification is not required except for these cables. The instruments served by these cables are: 2-IFI-310, 2-IFI-311, 2-IFI-320, 2-IFI-321, 2-CFI-419, 2-CLI-114, and 1-CLI-114.