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SUBJECT: Forwards requested info re post-implementation review of Category A requirements of NUREG-0578.

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March 10, 1980
AEP:NRC:00334B

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

On February 26, 1980, we met with the members of your staff performing the post-implementation review of the Category "A" requirements of NUREG-0578 for the Donald C. Cook Nuclear Plant Units 1 and 2. As a result of the meeting certain action items were generated, most of them requiring us to document what was presented to your staff during the meeting. At the conclusion of the meeting we were informed that a site visit by the review team would not be necessary and that certain other actions would be confirmed by the Cook Plant resident inspector. The attachment to this letter contains the information that was requested from us to complete the post-implementation review for Cook Plant.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

cc: R. C. Callen
G. Charnoff
R. S.Hunter
R. W. Jurgensen
D. V. Shaller -Bridgman

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ATTACHMENT

TO

AEP:NRC:00334B

This attachment presents American Electric Power's responses to requests for additional information and questions originated during the meeting held on February 26, 1980 with the NRC Staff Members performing the NUREG-0578, category A Items, post-implementation review for Cook Plant.

2.1.1

PRESSURIZER HEATERS

One block of pressurizer heaters with a 690 KW capacity can be connected to the 4KV bus 1A (2A for Unit 2) and one block of 690 KW can be connected to the 4KV bus 1C (2C). Bus 1A (2A) can be energized from emergency diesel generator 1AB (2AB) and bus 1C (2C) from emergency diesel generator 1CD (2CD).

The procedure for switching the heaters to emergency power supplies is written and in place at the Cook Plant. The procedure number is OHP 4023.001.014.

The switchgear which connects the pressurizer heater transformers to the 4KV buses is identical to the switchgear used for the diesel generator buses, and is installed to the same standards and is consistent with the design basis in the FSAR. The heaters are therefore energized through safety grade devices.

PORV & BLOCK VALVES

The present air system installed at Cook Plant for the PORV's consists of the normal control air system plus a supplemental system designed to meet the requirement for the low temperature reactor coolant system overpressure protection.

The plant control air system is described in FSAR Section 9.8.2. The supplemental system for the low temperature RCS overpressure protection consists of individual tanks connected to two of the three PORV's. These tanks are seismically supported, high pressure compressed air and/or nitrogen cylinders located in the upper containment with pressure reduction to the pressure level of the PORV actuator. Each cylinder,when full,contains adequate air to provide in excess of 170 valve operations.

The power sources for the solenoids of the three PORV's are:

NRV-151:Battery 1 AB (2AB)
NRV-152:Battery 1 AB (2AB)
NRV-153:Battery 1 CD (2CD)

The power sources for the three block valves are:

NMO-151:600 volt Bus 11A(21A)
NMO-152:600 volt Bus 11B(21B)
NMO-153:600 volt Bus 11D(21D)

The above are 250 volt and 600 volt safety grade buses. Bus 11D (21D) is fed from the Train CD diesel generator and buses 11A (21A) and 11B (21B) are fed from the Train AB diesel (See attached Figures 1 and 2).

PRESSURIZER LEVEL

The pressurizer level is measured by 3 wide range instruments NLP-151, 152 and 153:

NLP-151, fed from Channel I Inverter
NLP-152, fed from Channel II Inverter
NLP-153, fed from Channel III Inverter

Channel I and II Inverters are connected to battery 1CD (2CD).
Channel III Inverter is connected to battery 1AB (2AB).

2.1.3.a

DIRECT INDICATION OF VALVE POSITION

A separate annunciator is provided for each PORV which operates when the valve is not fully closed. The replacement limit switches for the PORV's will be seismically and environmentally qualified for containment accident conditions.

The acoustical monitoring system was purchased from Technology For Energy Corp. of Knoxville Tenn. The system provides the operator with an unambiguous indication of valve position. The acoustic system display is located in each of the Cook Plant control rooms. The system contains an independent alarm contact.

The acoustic system provides a reliable single channel direct indication and is powered from a vital instrument bus. The acoustic system is provided with the backup methods of individual safety valve discharge piping temperature indications and alarms, and temperature indications and alarm in the common discharge piping for the PORV's. These indications and alarms together with PORV limit switch indication are displayed to the operator in the control room. Our letter of May 1, 1979 (AEP:NRC:00185) in response to IE Bulletin 79-06A provides further information on the backup methods for determining valve position.

The system as stated by the equipment vendor and as installed meets seismic standards equal to or better than the seismic standards of the components to which it is attached.

The most limiting device from the environmental standpoint inside containment is a radiation-hardened preamplifier certified by the supplier for intermittent service at a temperature of 356°F. This device is being enclosed by a Crouse-Hinds enclosure to provide a vapor barrier which will be installed at the first outage of sufficient duration after equipment availability.

2.1.3.b

INSTRUMENTATION FOR INADEQUATE CORE COOLING

The present arrangement to use the existing plant computer was described in our letters of December 19, 1979 (AEP:NRC:00253B) and January 18, 1980 (AEP:NRC:00334). The computer calculates the margin to saturation for the hottest in-core thermocouple, the average of the thermocouples excluding the hottest and the coldest thermocouple; the hottest RTD and the average of RTD's again excluding the hottest and the coldest RTD. If any one of these four margins is less than the setpoint, an alarm is initiated.

The computer automatically uses the lower of two pressure inputs to calculate all margins to subcooling. The margin to subcooling is displayed on an analog trend device which is updated every four seconds.

The information provided in our letter AEP:NRC:00253B and its attachments regarding margins of uncertainty contained typographical errors. The overall uncertainty noted as "0" should have read "later". This information was provided in our letter of January 30, 1980 (AEP:NRC:00346) which discussed subcooling criteria for termination of safety injection. Simply stated, these uncertainties provide a 45°F margin if loop RTDs are used, or a 33°F margin if core exit thermocouples are used for temperature measurements.

The dedicated digital subcooling monitor is provided by Babcock & Wilcox. The sensor selection logic electronics are provided by Moore Industries Inc.

The subcooling meter input looks at eight (8) core exit T/C's or all eight (8) hot and cold leg RTDs and picks the single highest temperature from the group of temperature inputs selected. The pressure sensors and their associated components are qualified to the same seismic and environmental standards as other safety-related equipment of the same type and plant location. This also applies to the plant computer input sensors since the same sensor input is used. The subcooling meter inputs and the plant computer inputs are isolated from the engineered safety features and reactor protection function by appropriate current transformer isolators.

The subcooling meter as presently designed is qualified to the requirements of Regulatory Guide 1.97, Revision 2 draft.

2.1.4

CONTAINMENT ISOLATION

Each containment isolation valve is operated by its own control switch in the main control room. There are no cases where one switch operates more than one containment isolation valve.

2.1.6.a

SYSTEMS INTEGRITY FOR HIGH RADIOACTIVITY

The program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident was implemented on both Units of the Donald C. Cook Plant prior to January 31, 1980.

All liquid systems were tested by aligning each system per the applicable operating procedure and establishing normal operating pressure. Once these conditions were met, a visual inspection of the entire system was performed. Upon identification of any leakage, a suitable collection vessel was utilized to accumulate and measure leakage over a finite interval. This procedure was repeated and the average leak rate determined. This leakage was recorded as the "as found" leakage. Efforts to reduce this leakage were made and the measurements repeated, being recorded as "as left" leakage.

The following liquid systems were tested utilizing the above method with the leakages noted:

SYSTEM	AS FOUND LEAKAGE mL/HR		AS LEFT LEAKAGE mL/HR	
	Unit 1	Unit 2	Unit 1	Unit 2
CVCS - Letdown, Charg & RCP Seal Wtr.	0	11	0	0.3
Boron Injection	0	58	0	0
CVCS - Boron Holdup	0	0	0	0
ECCS - Safety Injection System	1.3	37	0	0.2
ECCS - Residual Heat Removal	5.5	0	0.36	0
ECCS - Containment Spray System	123	273	3.8	1.5
Liquid Waste Disposal System	0	0	0	0
Nuclear Sampling System	0	0	0	0

The gaseous systems were tested by first isolating the system or a section of the system to be tested and pressurizing that portion to approximately 2 psig with helium. All lines, valves and fittings were then traced utilizing a mass spectrometer to measure and record helium concentration. Appropriate corrective actions were taken as a leak was identified and repeat measurements then made. Leak rate was determined by utilizing the helium concentrations at the leak as well as within the test volume. All testing was performed with site personnel as well as representatives from Science Applications, Inc. and all portions of the waste gas system and gaseous sampling systems were included with test volumes terminating at containment isolation valves, where applicable.

The following gaseous systems were tested utilizing this method with the leak rates found as noted:

SYSTEM		AS FOUND LEAKAGE		AS LEFT LEAKAGE	
		Unit 1	Unit 2	Unit 1	Unit 2
Volume Control Tanks		0.3 - 2 cfm	None	3.5×10^{-5} cfm	None
CVCS Hold-up Tanks	for both Units 1 & 2 Combined	None		None	
Waste Gas Decay Tanks	for both Units 1 & 2 Combined	1.4×10^{-3} cfm		1.0×10^{-7} cfm	
Waste Gas Compressors	for both Units 1 & 2 Combined	None		None	
Vol. Control Tk. Gas Space Samp.		None	None	None	None
Cont. Hydrogen Monitoring		1.7×10^{-4} cfm	None	None	None

Table 1 lists those systems or portions of systems that were excluded from the liquid leak-testing requirements. Inability to use any of the excluded systems does not jeopardize any option for core cooling nor does it prevent the use of any needed safety systems.

During both the liquid and gaseous testing programs, a visual inspection of all included systems was performed and documented by marking system flow diagrams. This review, as well as documentation from the original pre-operational flushing programs ensuring that each section of installed piping was flushed as installed, provides adequate verification that all "as built" systems were installed as designed. Therefore, inadvertent releases due to construction errors in this piping are highly unlikely. In addition, the gaseous testing program included pressurization of many relief valve discharge lines, while those not included were physically verified by Operations personnel during testing.

Operating procedures exist with adequate valve line-up check lists for the transfer of radioactive liquids. Periodic review of these operating procedures since initial operation of both units has resulted in both procedure and design modifications to minimize the potential for inadvertent releases. Periodic audits of these procedures are performed by various groups and all procedures are reviewed and approved by the Plant Nuclear Safety Review Committee.

The above verifications, procedure reviews, installed leak detection systems as referenced in our January 18, 1980 submittal (AEP:NRC:00334), installed area and effluent monitors, and the Preventive Maintenance Program developed for leakage detection give adequate assurance that inadvertent releases as exemplified by the North Anna scenario are unlikely.

A Plant Manager Instruction was prepared to develop maintenance controls for leak reduction in the above systems. This procedure ensures a firm management commitment to maintaining liquid and gaseous leakage to a minimum, defines the responsibility for testing each of the systems at a frequency of once per refueling cycle, defines actions for maintaining test records, corrective actions for instances where leakage is noted as well as providing for a periodic review of the entire program's effectiveness. It should be noted that this program supplements and complements the normal, routine Job Order System and does not change requirements for normal inspection and surveillance.

2.1.6.b

PLANT SHIELDING REVIEW

In answering this question we follow the format of your checklist.

A. Design Review

1. A design review of the shielding of systems and equipment that would process radioactive fluids outside of the containment during an accident has been completed. The radiation doses have been calculated as a function of time for areas in the vicinity of these systems and for equipment using the primary coolant radioactive sources involved.

   Since the reactor vessel head will be modified to allow venting (remotely operated from the control room), use of the letdown system for non-condensible gas removal utilizing the CVCS and/or radwaste equipment located in the auxiliary building following an accident will not be required. The letdown system is isolated at the time of the accident by a Phase 'A' signal.

The containment sump water carried by portions of the engineered safety features (ESF) systems would be the only radioactive fluid that could be brought to the auxiliary building during an accident. In addition to this, the fluid contained in the post accident sampling lines will be brought to the auxiliary building. All required systems and supporting equipment that are vital for the safe shutdown of the reactor are considered functional during the first month following the accident. Hence, doses associated with equipment failures, maintenance, leakage other than normal, and airborne activity are not considered.

All the ESF and supporting systems can be operated from the control room not requiring access to the areas in which this equipment is located.

2. The TID source terms that were used in the shielding review are:

Noble Gases	100%	of core inventory
Halogens	50%	of core inventory
Others	1%	of core inventory

3. The General Design Criteria setforth in 10 CFR Part 50, Appendix A, Criteria 19, were used for allowable exposures.

4. All systems assumed to contain high levels of radioactivity are:

   Residual Heat Removal
   Containment Spray
   Safety Injection
   Charging
   Post Accident Sampling

5. All vital areas were identified and evaluated. As previously mentioned, the ESF and post-accident sampling systems are the only systems required to handle post-accident radioactivity outside the containment. All required equipment located in the containment and the auxiliary building can be remotely operated from the control rooms. Hence access to areas outside the control rooms for equipment operating purposes is not required except for post-LOCA sampling.

   Radioactive doses in the control rooms would be less than 1 mr/hr. Occupational radioactive doses in the on-site technical support center would be well within GDC 19. Special studies were made to determine the dose rates if access is required for the following instances:

   a) <u>Emergency Power</u>

   The diesel generators, battery room, switchgear room and ESF motor control centers requiring access, are all located in one section of the auxiliary building sufficiently shielded and away from the ESF system radioactive sources. This equipment is on three levels, with the diesel generators on the bottom level, below grade. Radiation doses in these equipment rooms would be well below 100

mr/hr. immediately following a LOCA with the diesel generator rooms less than 1 mr/hr. Since this equipment is not in a controlled access area, radiation protection measures would not be required. Motor control cabinets located in the controlled access area of the auxiliary building are not situated in the direct vicinity of the ESF equipment rooms. If access to this equipment were required, proper administrative measures for personnel entering the area would be taken and the total access time required to perform a local operation would result in a radiation dose within the limits of GDC 19.

b) Radwaste Control Panel

Operation of this equipment is not required as mentioned earlier. (A.1 above)

c) Recombiner Hookup and Control

This equipment, the hydrogen purge controls and the containment isolation valve reset controls are remotely operated from the control room and local operation is not expected.

d) Sampling Stations

e) Sampling Analysis Stations

The design criteria for the interim and permanent sampling stations and sampling analysis stations are such that the doses will be within the limits of GDC 19.

6&7. A response on equipment environmental qualification will be submitted to the NRC by April 15, 1980.

8. A version of the QAD code was used for this shielding review.

9. All radioactive sources were identified and as a minimum the following sources were considered:

a) Field-run piping

b) Post-accident sampling lines (normal sampling system would be isolated at the beginning of the accident).

c) All systems including their support equipment which might be needed for accident control as defined above.

d) Indirect radiation (skyshine) was considered.

e) CVCS was not considered as being required since venting of gas bubbles can be accomplished by venting through the reactor vessel head if required.

B. Modifications

1. Our review to date indicates that there is no need for modification.

2. Minor modifications, if required by the review under items A. 6 and 7, to protect radiosensitive components will be completed by 1/1/81.

3. There is no reason at present to believe that any modifications which may be required would not be adequate.

2.1.8.a

POST ACCIDENT SAMPLING

As described in Attachment 7 to our January 18, 1980 submittal, two interim systems have been installed to allow for post-accident sampling of the reactor coolant system and the containment atmosphere. All minor modifications are consistent with the safety class requirements of the associated system. Special procedures were written to utilize this equipment for taking samples, transporting them to the laboratory and preparing them for analysis. These procedures were approved by the Plant Nuclear Safety Review Committee by December 31, 1979 and all chemical personnel have had initial training in these procedures. Time studies have indicated that samples may be obtained within one hour after an accident using these systems.

Since these interim systems are merely extensions (to shielded areas) of original design sample points, it is felt that all samples will be representative of the associated system. The Reactor Coolant sample is intended to be an unpressurized sample and therefore, is assumed to be degassed and not adequate for dissolved gas analysis. Existing procedures have been modified to permit radioanalysis of both the coolant and containment atmosphere samples. Containment atmosphere hydrogen, coolant boron, pH and chloride will be determined utilizing existing procedures following dilution of the samples. All analyses are expected to be complete within one hour after obtaining the sample.

Additional portable shielding has been located in the laboratory to control exposures both during sample preparation and analysis. Sufficient precautions have been noted in sampling procedures and additional shielding provided at sampling stations to prevent overexposure to personnel.

Present shielding calculations indicate that the existing hot laboratory will be in the low mr/hr range following the accident and, therefore, functional during these activities. Background radiation levels in the counting room are estimated to be in the low mr/hr range and sufficient for counting all samples. In the event airborne levels become a problem in the existing counting room, an emergency counting facility was established in the Post 2 security building, utilizing a NaI detector/multi-channel analyzer configuration calibrated for all post-accident sample geometries.

As mentioned in previous submittals, a contractor, the NUS Corporation, has been asked to address the major modifications required by January 1, 1981. The present intent of these modifications is to include direct, in-line sampling of the reactor coolant, containment atmosphere and provisions for containment sump sampling. As the intent is to utilize modified existing sampling where possible, sampling and analysis within the two hour limit is considered readily

achievable. See our letter of January 31, 1980 (AEP:NRC:00334A) for further information on the conceptual modifications.

In-line sampling will be utilized for coolant boron, chloride, pH, hydrogen and possibly dissolved oxygen while radioanalysis will be performed on a diluted sample. Similar arrangements are planned for the containment atmosphere samples.

Counting and analysis facilities are still under study although the facilities described for the minor modification/interim program appear adequate. Locations for the sample stations have been proposed by our contractor and these are still under review by AEP. Preliminary shielding studies indicate that personnel overexposure will not be a problem.

2.1.8.b HIGH RANGE EFFLUENT MONITOR

Before returning the unit to service, a plant modification was completed to the plant vent sampling system which facilitates the use of an interim method for determining plant releases should existing monitors exceed their upper limits. Since the unit vent is the sole release point for all release paths from the auxiliary building only that sampling system was modified.

The modification mentioned above consisted of extensions of the vent sampling line for each unit along with remotely operated control valves and pump. A three foot section of the line is brought through the auxiliary building wall and shielded with four inches of lead. A removable lead plug permits introduction of a portable radiation monitor in a reproducible position, viewing the line from a distance six inches from the centerline. The location of this shielded section of piping was selected to permit the most convenient access with minimum exposure potential.

Procedure 12 THP.SP.012 approved on December 31, 1979 describes the location and operation of the remotely operated valves and pump as well as the use of the shielded segment of line. It also gives decisional aids to allow rapid calculation of either concentration (range of 1 μCi/cc to 7 x 10^4 μCi/cc) or release rate (upper range of 5 x 10^5 to 1.6 x 10^6 Ci/sec depending on vent flow rate).

The monitoring system is designed to be used with one of two portable dose rate monitors, both of which are in use at the plant. For radiation levels less than 2 R/hr at the monitoring window (1 x 10^3 μCi/cc or about 2 x 10^4 Ci/sec at 70,000 cfm vent flow), an RM-16 area monitor with HP250 probe would be used because the detector can be separated from the electronics readout package which could then be located remotely in the lowest radiation level possible. Should this equipment not be appropriate or available, a range PIC-6A portable ion chamber instrument (upper range of 1000 R/hr) can also be used. Procedures for operation (12 THP 6010.RAD.536 for RM-16 and 12 THP 6010.RAD.518 for the PIC-6A) and calibration (12 THP 6010.RAD.586 and 12 THP 6010.RAD.568 respectively) have been approved and in use for a number of years. The PIC-6A reads essentially linearly (with respect to gamma energy) for all photon emergencies above 50 KeV. The HP-250 detector reads high between 70 and 300 KeV (reaching a peak of two times actual exposure rate at 100 KeV) and essentially linearly above 300 KeV. The RM-16 can be power by available AC power or, that being unavailable, by the internal batteries which have a working lifetime of 35 hours without charging. The PIC - 6A uses easily replaceable batteries which have an average lifetime of 60 hours. Either should provide adequate life for more than 7 days before battery replacement or recharge would be necessary.

Releases from the Turbine Building (steam dump and SJAE) will be estimated utilizing real time release rates obtained from portable area monitors. Permanently installed monitors will be made available for the potential release points in the Turbine Building by January 1, 1981.

All Radiation Protection personnel were trained in the use of the monitoring system (Procedure 12 THP.SP.012) prior to January 1, 1980. The portable instruments are part of the normal plant equipment so no special training was required.

2.1.8.c IMPROVED IODINE INSTRUMENTATION

By December 31, 1979, we had available the following portable equipment capable of monitoring for radioiodine in the presence of noble gases with the appropriate operation and calibration procedures indicated:

2 - Nuclear Measurement Corporation continuous air monitors (12 THP.6010.RAD.031 and 12 THP 6010.RAD.581) using single channel analysis without noble gas subtraction.

4 - Eberline Instrument Corporation PINE-1A monitors (12 THP 6010.RAD. 532 and 12 THP 6010.RAD.582) using single channel analysis with Xenon subtraction capabilities.

All detectors are in 3 inch lead shields and are capable of using silver Zeolite sample cartridges which were available on-site prior to January 1, 1980. The two NMC monitors are stocked with only Ag-zeolite sample cartridges and are located (and dedicated), one for monitoring either Control Room and one for monitoring the Temporary Technical Support Center. The Eberline units are located on various levels of the auxiliary building. They are normally operated with TEDA impregnated charcoal but may also be used with Ag-zeolite when necessary. Being portable they may be used wherever it might be deemed most appropriate.

In addition, numerous portable air samplers are available (either AC or 12V DC powered) for grab or continuous sampling and can be used with either TEDA charcoal (routine operation) or Ag-zeolite as necessary. Counting of these iodine samples, as well as analysis of samples from the above mentioned portable monitors may be performed using normal laboratory analysis equipment, the equipment in the low-background facility, or using an available Eberline MS-2 portable single channel analyzer which has been fitted with a 2" x 2" NaI crystal in a 2¼" lead shield designed for counting iodine samples.

Procedures both for operation and calibration are available for either type of air sampler (12 THP 6010.RAD.206/256 for AC powered and 12 THP 6010.RAD.210/260 for DC powered) and the MS-2 (12 THP 6010.RAD. 525/575). No special training was required for the use of the air samplers as they are routinely used equipment. Special training was given to all Radiation Protection personnel prior to December 31, 1979 for use of the Eberline Monitors, and NMC monitors, the MS-2 single channel analyzer, and the use of Ag-zeolite for special sampling. Also presented were the contents of a special standing order, TSO.018, which identifies those procedures which would be of special value and use during emergency conditions.

2.1.9

RCS VENTING

This response supplements the description given in our letter No. AEP:NRC:00253D, dated January 15, 1980 which described the design of the RCS high point vents. Remote-operated vents are provided for the reactor vessel head and the pressurizer steam space. Procedures will be developed which will detail the conditions for using and not using the vents, taking into account the effects on coolant inventory, core cooling and containment integrity.

The reactor vessel head vent and the pressurizer vent each consist of two parallel paths. Each path has two power-operated, fail-closed valves in series. Each path has a restriction orifice at the inlet to limit flow. When venting hydrogen from either vessel at a pressure of 2400 psia and at temperature of 600^{o}F the capacity of one path is approximately 5795 cu. ft. per hour which is 46% of the total RCS volume.

The inlet orifice size limits the flow to less than the small LOCA definition for Cook Plant in the event of inadvertent opening. In addition, each path has two valves in series which are controlled independently.

Each valve has indicating lights on the main control panel to indicate the open and closed position of the valve. There is a temperature indication for the discharge piping to identify valve leakage.

Each valve has a control switch located on the main control panel.

The valves, piping, and supports will be qualified to the appropriate seismic requirements for Cook Plant.

The vents connect to the existing reactor head vent and to an existing vent connection on the pressurizer safety valve inlet piping as shown in Figures 1 & 2 of our January 15, 1980 (AEP:NRC:00253D) submittal.

The equipment will be qualified to the same or higher requirements as for other reactor protection equipment in a similar location. Each vent (two valves in series) will be powered from one station battery. The parallel vent will be powered from the other station battery. These are safety grade power supplies.

The power will be removed, except when venting is required, to prevent inadvertent operation of the valves.