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December 31, 1980

1-120-17
2-120-31

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Nuclear One - Units 1 & 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6
NUREG-0737
(File: 1510.6, 2-1510.6)

Gentlemen:

Attached are the submittals required by your letter of October 31, 1980. These submittals are made in accordance with AP&L's letter dated December 19, 1980, which provided commitments to the subject requirements.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

DCT:MAS:sl

Attachment

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I.A.1.1 SHIFT TECHNICAL ADVISOR

NUREG-0737 and the H. R. Denton letter dated October 30, 1979, state the requirements for STA education and training. These requirements and our responses are as follows:

I. The Shift Technical Advisor shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline,

Response:

All ANO STA's have a Bachelor of Science Degree in engineering or engineering technology.

II. and have received specific training in the response and analysis of the plant for transients and accidents.

Response:

The training which addresses this requirement was conducted as portions of A-1, B-1, B-3, C-1 and C-2 from the STA Training Program described below.

III. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room.

Response:

The training which addresses this requirement was conducted as portions of A-1, B-1, B-2 and B-3 from the STA Training Program described below.

LONG TERM STA TRAINING WILL INCLUDE THE FOLLOWING:

I. Initial Training

New STA's with a degree or equivalent in an engineering or scientific discipline will complete the initial STA training program.

II. Requalification Training

Each year STA's will receive as a minimum 40 hours of training on a simulator for each unit. This 40 hours will include 20 hours of classroom training and 20 hours of training on the simulator.

In addition, STA's will receive a minimum of 40 hours of training in selected topics including transients and accidents.

SHIFT TECHNICAL ADVISOR TRAINING PROGRAM

PURPOSE:

Provide a formal training program for the STA position, based on identified training objectives, to insure that the STA has the technical knowledge and operating experience to evaluate plant conditions (normal and accident) to enhance safe operation of the plant.

OBJECTIVE:

Provide the specific training to the STA, in addition to a Bachelor's Degree in a science or engineering discipline, to insure the STA has technical and analytical capability to support the diagnosis of off-normal events and advise the Shift Supervisor on actions to terminate or mitigate the consequences of such events. The general areas of training that have been identified to develop these capabilities are listed below.

A. Plant Structures, Systems, Component Design and Layout

1. Lecture series covering systems and components for both units.

UNIT ONE

a. Reactor Coolant System
b. Core Construction
c. Makeup and Purification System
d. Emergency Core Cooling Systems
e. Once Through Steam Generators
f. Emergency Feedwater System
g. Electrical Distribution System
h. Emergency Diesel Generator System
i. Service Water System
j. Intermediate Cooling Water System
k. Secondary Systems

UNIT TWO

a. Reactor Coolant System
b. Core Construction
c. Chemical and Volume Control System
d. Safety Injection Systems
e. Containment Spray System
f. Electrical Distribution
g. Emergency Diesel Generator System
h. Secondary Systems
i. Emergency Feedwater System

2. An on-the-job training (OJT) program requiring completion of Plant Systems and Control Room Training Guides for both units.

a. Operating and Emergency Procedures
b. Technical Specifications

Selected sections of the Plant Systems Guides were completed by December 31, 1980.

B. Functions and Capabilities of Instrumentation and Controls in the Control Room

1. Lecture series covering instrumentation and controls of both units.

UNIT ONE

a. Non-Nuclear Instrumentation
b. Engineered Safeguards Actuation System
c. Reactor Protection System
d. Nuclear Instrumentation
e. Incore Instrumentation
f. Control Rod Drive System
g. Integrated Control System
h. Steam Line Break Instrument and Control System
i. Area Radiation and Process Monitors

UNIT TWO

a. Primary Plant Instrumentation
b. Reactor Protection System
c. Engineered Safety Features Actuation System
d. Reactor Regulating System
e. Steam Dump and Bypass System
f. Feedwater Control System
g. Nuclear Instrumentation
h. Core Operating Limit Supervisory System
i. Megawatt Demand Setter
j. Core Protection Calculator
k. Radiation Monitoring System

2. An OJT program requiring completion of Control Room Training Guides for both units. These guides were completed by December 31, 1980.

3. Shift Technical Advisor simulator training.

C. Plant Response and Analysis for Transients and Accidents

1. Lecture series including

a. Reactor Theory
b. Thermodynamics and Core Hydraulic Considerations
c. Plant Transients
d. Safety Analysis

2. Shift Technical Advisor simulator training.

D. Written Examination Covering the STA Training Program

I.C.1 PROCEDURES FOR TRANSIENTS AND ACCIDENTS

ANO-1:

Inadequate Core Cooling guidelines were submitted by AP&L letter dated December 13, 1979. These guidelines were incorporated into existing small break LOCA guidelines and into appropriate operating procedures. AP&L has received no comment on these guidelines from the NRC staff. AP&L is therefore in full compliance with NRC requirements relating to ICC guidelines. NRC concerns will be addressed, and guidelines re-analyzed as necessary, after staff review and comment on the guidelines.

Guidelines for development of procedures for transients and accidents are being developed by AP&L and other B&W owners as part of the ATOG program. As stated in our letter of December 19, 1980, the NRC staff has been appraised of the ATOG program since its inception in mid-1979. A meeting was held on December 16, 1980 between AP&L and other B&W owners and NRC staff members to discuss this subject. It was decided in this meeting that the draft guidance on ATOG which was issued in August, 1980 was sufficient to meet the January 1, 1981 requirements.

ANO-2:

Inadequate Core Cooling (ICC) guidelines were developed and documented in CEN-117 dated October, 1979, and appropriate procedure revisions have been made by AP&L. Review and re-analysis of these guidelines will be performed, as necessary, after receipt by AP&L of specific NRC staff concerns.

AP&L is presently participating with other CE owners to produce guidelines for development of procedures for transients and accidents. A discussion of this program was provided in our December 19, 1980 letter. Also, a meeting between the CE owners and the NRC staff on this subject is scheduled for January, 1981. The purpose of this meeting will be to discuss the submittal of revised emergency procedure guidelines, currently scheduled to be submitted for review to the NRC staff during the summer of 1981.

II.B.2 PLANT SHIELDING

ANO-1 and 2:

1. Introduction

   Section 2.1.6.b of NUREG 0578 recommended a review of the adequacy of nuclear power plant shielding to allow recovery from an accident. This review was performed prior to January 1, 1980. The following summarizes the results of these reviews and addresses action being taken to assure the capability of the operators to control and mitigate the consequences of an accident.

2. Source Terms

In an attempt to quantify maximum possible radiation source levels, it was assumed that the core had reached the steady-state fission product inventory defined in our FSAR's and that 100% of all noble gases (Krypton and Xenon), 50% of all iodine and 1% of all remaining radioactive particulates were dissolved in the primary coolant. Reactor coolant taken from the sump was assumed to contain 0.1% of the noble gases, 50% of the iodine and 1% of the remaining fission products. For systems containing containment atmosphere, 100% of the noble gases and 25% of the core iodine were assumed to be mixed in the containment atmosphere.

3. Systems Containing the Source

Time dependent source terms, based on component volume and geometry, have been established for components of the following systems:

Letdown and Makeup
Decay Heat Removal
Reactor Building Spray
Safety Injection
Liquid and Gaseous Radwaste
Sampling

4. Vital Areas for Plant Recovery

During an accident, damages are assessed by the operators. If operations outside the control room are necessary, operators accompanied by health physics personnel are dispatched. All areas of the plant are considered equally vital. Our review provides guidance as to where excessive radiation levels may exist after inadequate core cooling accidents that lead to significant core damage. If needed, the Technical Support Center and the Post-Accident Sampling Facility (PASS, under construction) will be manned. Sample analysis will be accomplished in the sampling facility. The Technical Support Center is located in the Administration Building where individual doses averaged over 30 days will be very small for all accident conditions. By January 1, 1982, the new sampling and sample analysis facility (PASS) will be operational. Shielding for this building has been designed to allow post-accident occupancy. Control room dose rates may exceed GDC 19 criteria after core degradation has occurred if letdown and RCP seal return flow continues, because the Makeup Tank for ANO-1 and the Volume Control Tank for ANO-2 may develop very high radiation levels. Since this is a result of inadequate core cooling, the Inadequate Core Cooling Operator Guidelines are being reviewed to determine the most appropriate operator response to this situation.

5. Dose Rate Maps

Construction of realistic dose rate maps requires detailed knowledge of the accident scenario which the map represents.

The development of overly conservative dose rate maps representing unrealistic accident scenarios serves no useful purpose. Since the expense for preparation of such maps cannot be justified, they will not be prepared.

II.E.1.2 EFW AU OMATIC INITIATION AND FLOW INDICATION

ANO-1:

The necessary information was provided by AP&L letters dated October 15, 1980 and December 3, 1980.

ANO-2:

No modifications are necessary.

II.E.3.1 EMERGENCY POWER FOR PRESSURIZER HEATERS

ANO-1:

As a result of NUREG-0578, item 2.1.1, AP&L installed an additional 42KW of pressurizer heater capacity. The pressurizer heaters, which are powered from the Engineered Safeguards (ES) buses, are now capable of providing 126KW of pressurizer heater capacity from either ES bus. The additional heaters are powered from the 480 volt swing bus using a qualified combination starter of the circuit breaker type. This circuit is controlled from the control room by means of an on-off switch and is not tripped on ESAS but on pressurizer low level. This design is consistent with similar applications at ANO-1. It also should be noted that ANO-1 was not designed to meet Reg. Guide 1.75.

ANO-2:

The ANO-2 proportional heaters are powered from the Engineered Safety Features (ESF) buses. An SIAS trip was added during the Millstone modification and was removed per item 2.1.1 of NUREG-0578. This fix enables the operator to manually load the pressurizer heaters on to the emergency power source. A change over from offsite power to on-site power will result in a trip of the heaters. Manual reinitiation is then required.

The pressurizer heater power supply arrangements for both ANO-1 and 2 were reviewed at the plant site in early 1980 by an NRC Lessons Learned task force inspection team. The designs were found to be acceptable as documented in the Safety Evaluation Reports issued on March 10, 1980.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

ANO-1 and 2:

As stated in AP&L letter dated December 19, 1980, design and qualification criteria for the existing incore and core exit thermocouples, which are used to detect ICC, will be provided after determination by the NRC staff that use of this instrumentation is an acceptable method of detecting ICC.

II.K.2.10 ANTICIPATORY REACTOR TRIP SYSTEM

ANO-1:

Design information was provided by AP&L letter dated August 8, 1980.

II.K.2.13 THERMAL SHOCK REPORT

ANO-1:

This report is being submitted under separate cover.

II.K.3.2 REPORT ON PORV FAILURES

ANO-1:

AP&L has just recently completed this analysis. The report will be submitted by January 8, 1981 after it receives proper review.

ANO-2:

ANO-2 has no PORV.

II.K.3.3 REPORT ON SV & RV CHALLENGES

ANO-1 and 2:

This report will be submitted under separate cover by January 8, 1981.

II.K.3.7 EVALUATION OF PORV OPENING PROBABILITY

ANO-1:

This evaluation was performed as part of the analysis required by II.K.3.2, and will be submitted as part of that report.

ANO-2:

ANO-2 has no PORV.

II.K.3.17 ECCS OUTAGES

ANO-1 and 2:

This information will be provided ' · March 1, 1981, as committed by AP&L letter dated June 18, 1980.

II.A.2 EMERGENCY PREPAREDNESS

ANO-1 and 2:

This information will be submitted under separate cover.

III.D.3.3 IODINE INSTRUMENTATION

ANO-1 and 2:

AP&L is presently in compliance with this item. Two portable detectors complete with single-channel analyzers are onsite at ANO, one located in the Technical Support Center and one in the Control Room. These detectors are capable of accurately monitoring and measuring iodine. This equipment was specifically recommended by an NRC Lessons Learned task force inspection team which visited the ANO site in early 1980.

III.D.3.4 CONTROL ROOM HABITABILITY

ANO-1 and 2:

The present designs of the ANO-1 and 2 control rooms meet the requirements of General Design Criteria 19, as stated in section 9.5.1 of the ANO-2 SER. Therefore, no modifications to the control rooms are required. The required information will be submitted by March 15, 1980.