EVALUATION OF LICENSEE'S COMPLIANCE WITH
CATEGORY "A" ITEMS OF NRC RECOMMENDATIONS
RESULTING FROM TMI-2 LESSONS LEARNED

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ARKANSAS NUCLEAR ONE - UNIT 2

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Introduction

I. INTRODUCTION

By letters dated October 17[1], November 29[2], December 5[3] and 17[4], 1979 and January 17[5], 18[6], 29[7] and 31[8], and February 29[9], 1980 Arkansas Power and Light Company (licensee) submitted commitments and documentation of actions taken at Arkansas Nuclear One, Unit 2 (ANO-2) to implement our requirements resulting from TMI-2 Lessons Learned. To expedite our review of the licensee's actions, members of the staff visited the licensee's facility on January 21, 22 and 23, 1980. This report is an evaluation of the licensee's efforts to implement each Category "A" item which was to have been completed by January 1980. The items will be completed prior to startup from the current shutdown, except as noted.

II. EVALUATION

Each of the Category "A" requirements applicable to PWRs is identified below. The staff's requirements are set forth in Reference 10; the acceptance criteria is documented in Reference 11. The numbered designation of each item is consistent with the identifications used in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Heaters)

The licensee has determined that at least 150 kilowatts of pressurizer heaters should be available from an assured power source within 30 minutes of loss of offsite power to maintain natural circulation. The required heater capacity is based on pressurizer ambient losses measured during startup testing. We find the determination of adequate heater capacity to be acceptable.

The design has available 150 kilowatts of proportional heaters on each safety bus. The existing control circuit for these heaters will be modified to allow closing of the power circuit breakers upon loss of offsite power following safety injection actuation signal. Upon loss of offsite power, these heater breakers will be tripped by the under voltage signal and will be manually loaded on the diesel generator. This can be accomplished from the control room. This is in conformance with our requirements outlined in NUREG-0578 and is acceptable. Our Office of Inspection and Enforcement will (1) verify the installation of this design feature and (2) the adequacy of the procedures to guide the operator to put these heaters on the safety buses within 30 minutes are included in the operating procedures. This will be documented in an appropriate inspection report.

EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Level and Relief and Block Valves)

The ANO-2 design does not incorporate a PORV or block valve. The pressurizer level instrumentation is safety-grade, redundant and powered from the safety buses as a feature of the approved existing design.

2.1.2 RELIEF AND SAFETY VALVE TESTING

The licensee has stated in his response to this item that it will participate in the EPRI program to conduct performance testing of PWR relief and safety valves.

A description of the test program was provided by EPRI in December 1979. At present, this program is under NRC review to ensure that the NUREG-0578 requirements are met.

We believe that this commitment provides assurance that the requirements for performance testing of relief and safety valves will be satisfied. The basis for this acceptance is that we will review the test program to confirm the applicability to ANO-2. Completion of the test program is on a schedule different from Category "A" items. Therefore, we conclude that the licensee has satisfied the Category "A" requirements of this item.

2.1.3.a DIRECT INDICATION OF PORV AND SAFETY VALVE POSITION

To meet this NUREG-0578 requirement the licensee has installed an acoustical monitoring system to monitor the position of the safety valves. The acoustical system consists of a hermetrically sealed piezoelectric sensor mounted immediately down stream of the safety valve piping. Redundant sensors are provided for each valve. The sensors are connected to the preamplifiers (one for each sensor) which are located inside containment and transmit redundant signals to a signal conditioner located in the control room. The acoustical valve position indication equipment is powered from a class IE bus. All equipment will also be mounted as seismic class IE installations. In addition, the equipment is environmentally suited for its application. The licensee has stated that sufficient qualification documentation is not available at present, however, generic qualification should begin in February 1980, and will require six months to complete.

Backup valve position indication is provided from temperature elements located downstream of the safety valves. These are monitored in the control room and alarm on high temperature.

Based on our review of the licensee's design, we conclude that the licensee is in compliance with the direct indication of power operated relief valve and safety valve position and schedule requirements as outlined in NUREG-0578, and is, therefore, acceptable. Our Office of Inspection and Enforcement will verify (1) installation of the above design, (2) that the procedures to use backup valve position indication are included in the operating procedures, and (3) the adequacy of the qualification documentation of this equipment when made available. This will be documented in appropriate inspection reports.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (Existing Instrumentation)

The licensee has stated that its emergency procedures and associated operator training with regard to existing instrumentation to be used by the operator to detect inadequate core cooling conditions, have been updated based on guidelines provided by CE report CEN-117. This report was submitted separately in response to NRC I&E Bulletin 79-06C. Our evaluation thereof for this item will be reported separately.

ADDITIONAL INSTRUMENTATION

The licensee has looked at several conceptual designs for reactor vessel water level indication from C-E as part of the C-E Owner's Group effort. We conclude that the licensee has satisfied our short term requirement. However, the need to supplement existing instrumentation and to provide additional unambiguous indications of inadequate core cooling are still under review. We will complete this item during the review of Category B items.

SUBCOOLING METER

The licensee will install two primary coolant saturation meters. These saturation meters will continuously display the margin between actual primary coolant temperature and the saturation temperature (the temperature at which boiling occurs).

Two channels of saturation margin measurement and indication will be provided. Each channel consists of a calculator and a digital display. The pressure input for each calculator will be derived from safety grade, wide-range (0-3000 psig) pressurizer pressure transmitters. The temperature inputs for each calculator will be from redundant, safety-grade wide-range (150-750°F) T_{hot} RTD's in each loop.

The temperature margin to saturation conditions from each calculator is continuously recorded on a two channel strip chart recorder located in the control room. This condition can also be individually read from digital indicators mounted in cabinets located in the control room. Annunciation of low margin to saturation is provided in the control room.

In addition to the above redundant saturation meters, backup capability already exists to detect inadequate core cooling conditions from the plant computer. In addition, primary coolant temperature and pressure are directly available to the operators by means of existing console indicators. Steam tables are also provided in the control room for use by the operators to determine saturation margin manually.

Based on our review of licensee's design, we conclude that the design of the subcooling meters is in compliance with our requirements as outlined in NUREG-0578 and is acceptable. Our Office of Inspection and Enforcement will verify the implementation of this design and the proper qualification documents of the meters, calculators and cabinets. This will be documented in an appropriate inspection report.

2.1.4 CONTAINMENT ISOLATION

The licensee has modified the containment isolation provisions so that diverse par[illegible]s will be sensed to assure automatic isolation of non-essential systems under postulated accident conditions. The parameters to be used are containment high pressure (CIAS) or low reactor coolant system pressure (SIAS). The licensee has identified 21 systems in four categories that have automatically actuated valves which provide penetration isolation. All of these systems isolate automatically on CIAS signal. Category I (which has only chemical and volume control system) by original design will isolate on either CIAS or SIAS. Category II valves isolate only on CIAS signal. Categories III and IV are being modified so they isolate on either CIAS or SIAS.

The licensee has provided the following justification for not providing diverse signals to Category II systems.

All systems listed in Category II are closed loop cooling systems. These are normally open during power operation and are necessary for "normal" orderly cooldown. In addition, none of these systems have direct contact with the reactor coolant such that non-isolation would allow contaminated fluid to escape unless the integrity of the closed loop system was also violated. We find this justification acceptable. Items in Category III are normally closed during power operation and require specific manual operation for opening. To provide for an increased margin of safety, the licensee will provide a low reactor coolant pressure signal to isolate in case these system valves were inadvertently open. Items in Category IV are normally open during power operation and require specific manual operation to close them. For these systems, the licensee will provide a low reactor coolant pressure signal to isolate these systems. We find these justifications acceptable.

The present design does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Moreover, reopening of containment isolation require deliberate operator action. The licensee has the capability to reopen only a selected number of valves after they are closed on containment isolation. It should be noted that if the containment isolation signal clears then the design has the capability to permit manual re-opening, after resetting, any of the isolated valves.

We conclude that the licensee has satisfied our short term requirements with regard to containment isolation, as set forth in NUREG-0578. Our Office of Inspection and Enforcement will verify the installation of this design and document this in an appropriate inspection report.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

ANO-2 was licensed to use recombiners inside containment for post-accident combustible gas control of the containment atmosphere. There are no penetrations of the containment for these recombiners and none are needed. Based on this, we conclude that the licensee has met both the Category "A" and Category "B" requirements for this Lessons Learned item. No further action needs to be taken.

2.1.5.b INERTING BWR CONTAINMENTS

This short-term Lessons Learned item does not apply to ANO-2.

2.1.5.c RECOMBINER PROCEDURES

The licensee has reviewed the procedures and shielding for operating the system at ANO-2 which provide post-accident combustible gas control of the containment atmosphere during an accident. The system is redundant, safety-grade recombiners inside containment. The licensee has concluded that no changes are needed to the procedures and shielding for this system.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements. There are no Category "B" requirements. Verification of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.6.a SYSTEM INTEGRITY

The licensee has listed the plant systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. These systems are the dirty liquid radwaste, gas radwaste. chemical and volume control, decay heat system (low pressure safety injection) and reactor building spray system. High pressure safety injection is included in the volume control system. The licensee has implemented an immediate leak reduction program for these systems to reduce their present leakage. The licensee has reported the measured "as found" leakage and the "as reduced" leakage for these systems in ANO-2 to NRC.

The licensee has established and implemented a permanent leak reduction program to keep future leakage from these systems to as-low-as-practical levels. This program includes integrated leak rate tests once per refueling cycle; identification of leakage from visual surveillance by plant personnel, area radiation monitors and effluent monitors and corrective actions taken; and the existing plant preventative maintenance program.

The licensee has also reviewed the plant design for potential leakage release paths from the above system due to design and operator deficiencies as discussed in an NRR letter to the licensee regarding North Anna and Related Incidents dated October 17, 1979. No corrective actions by the licensee were needed.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item. There are no Category "B" requirements. Verification of the procedures which implement the licensee's permanent leak reduction program will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.6.b PLANT SHIELDING REVIEW

The licensee has performed a radiation and shielding design review of the spaces around the plant systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. These systems are listed in the evaluation of item 2.1.6.a. This design review has been provided to NRC which includes maps of the estimated dose rates in different areas of the plants. The radioactive source terms assumed for the design review are consistent with the source terms given in the NRC clarification letter dated October 30, 1979. The licensee will propose long-term changes, if any are needed, to NRC by March 31, 1980. For the short-term, the licensee will provide portable shielding onsite for use in areas to reduce personnel exposure.

The licensee did identify the location of vital areas and equipment and instrument areas in which personnel occupancy may be limited. These areas are the sample room, decay heat pump room (i.e., valves must be manually operated), and certain areas of the control room. Corrective actions for the long term, beyond 1980, to provide adequate access to the control room and sampling area will be provided in the shielding report. We discussed with the licensee during our plant trip corrective actions for the long term, beyond 1980, for the decay heat pump room to not require access to this room during an accident. The current thinking to correct this problem is to replace certain manual valves by motor-operated valves in this room. The licensee will identify any areas in which safety equipment may be unduly degraded by radiation fields during post-accident operations in the shielding report.

The major contributor to the radiation fields is the chemical and volume control system. The licensee has concluded for this year that the levels of radioactivity in the chemical and volume control system can be limited with existing plant equipment and procedures.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements. An evaluation of the above design review and the licensee's corrective actions will be performed as part of the review of the Category "B" Lessons Learned requirements.

2.1.7.a AUTO INITIATION OF AUXILIARY FEEDWATER SYSTEMS

This item is being reviewed separately by NRC and our findings will be reported in a separate report.

2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS

The licensee has installed, as part of the original design, flow indication on each of the flow paths to the two steam generators. This arrangement meets the Category "A" requirements and is acceptable. The licensee will need to upgrade the system to include redundant, safety-grade indication to each of the steam generators as a Category "B" requirement.

2.1.8.a POST-ACCIDENT SAMPLING

The licensee has performed a design and operational review of the reactor coolant and containment atmosphere sampling. The licensee has provided procedures to provide the capability in 1980 to promptly take a reactor coolant sample during a serious transient or accident and to minimize personnel radiation exposure. The licensee has provided interim procedures to take a containment air samples in 1980 in an area to minimize personnel exposure. In addition, portable shielding will be available onsite for use in the plant reactor coolant sampling room and analysis facilities to provide a capability to take a sample and to analyze it during an accident. A modified sampling system to minimize personnel exposure will be proposed in the March 31, 1980 shielding report. New reactor coolant system sampling points will be included in this proposal to allow collection of a representative sample without operating the letdown into the chemical and volume control system. Any long term changes which are needed for the plant analysis facilities will be proposed in the shielding report.

The licensee has performed a design and operational review of the plant radiological analysis facility. This facility has been moved to an area just off the turbine deck adjacent to the control room to reduce the possible radiation background in this facility during an accident. Procedures are available to provide the capability to promptly quantify radionuclides in a highly radioactive sample during a serious transient or accident.

The licensee has performed a design and operational review of the plant chemical analysis facility. Procedures are available to provide the capability to promptly quantify certain chemical analyses in a highly radioactive sample during a serious transient or accident.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements. Verification of the procedures for taking and analyzing a reactor coolant and containment sample will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.8.b HIGH RANGE EFFLUENT MONITORS

The licensee has provided an interim method for quantifying high level noble gas effluent from the Auxiliary Building ventilation line. This line and the secondary side steam dump line are the only ones used during a serious transient or accident involving the reactor. The licensee is studying how to monitor gross radioactivity releases from the plant steam dump valves. The licensee will provide a design and install a radiation monitoring system on this line and provide interim procedures to use this system to quantify noble gas releases from this line by March 31, 1980.

The licensee has installed a new iodine/particulate sampling system for the above lines to provide sampling at a more accessible location. The iodine/particulates are collected on cartridges and taken to the plant radiological counting facility for analysis. Procedures have been developed for collecting and analyzing these samples.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements. Verifications of the procedures for quantifying high-level noble gas and iodine/particulate effluents from the plant will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.8.c IMPROVED IODINE INSTRUMENTATION

The licensee has equipment which can be dedicated to analyzing air samples for radioiodine concentrations during an accident. The licensee states that sample collection and counting times can be minimized, in an accident situation, so that a rapid analysis can be made to determine if significant inplant airborne radioiodine concentrations are present. The licensee has also ordered two single channel analyzers which can be used to promptly and accurately analyze air samples for airborne radioiodine during an accident. These analyzers will be located onsite: one in the ANO-1/ANO-2 control room area and one in the Technical Support Center where plant personnel will be stationed during an accident. The analyzers will be available by May 1, 1980.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements. There are no Category "B" requirements. Verification of the procedures that state the licensee has equipment dedicated to analyzing air samples during an accident, that the above equipment is in place in the ANO-1/ANO-2 control room area and in the Technical Support Center, and is periodically checked and calibrated will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.a SHIFT SUPERVISOR RESPONSIBILITIES

The licensee has revised plant procedures, as necessary, to set forth the responsibilities of the Shift Supervisor such that he can provide direct, command oversight of operations and perform management review of ongoing operations that are important to safety and not be distracted from these important responsibilities by administrative details.

We conclude that the licensee has satisfied the Category "A" requirements to provide revised responsibilities and authority for the Shift Supervisor. Verification of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.1.b SHIFT TECHNICAL ADVISOR

The licensee has provided an on-shift technical advisor (STA) to the shift supervisor to fulfill the two required functions of accident assessment and operating experience assessment. The STA should be able to report to the control room within 10 minutes to assist in diagnosing an off-normal event.

For the interim period of 1980 the licensee has provided additional site engineering staff personnel (i.e., degreed engineers) to be the STA. They will serve a 24-hour duty day on a rotating basis and will be onsite at all times during their duty. The same person will be the STA for both ANO-1 and ANO-2. Sleeping quarters are available onsite and the STA will be available to the control room within 10 miles of being called. In the event of an accident, the designated STA is removed from the chain of command. He then acts only as an advisor to the Shift Supervisor who is in charge of all actions on the affected unit. The operational experience analysis is the primary responsibility by the on-site Plant Performance group. The designated STAs are responsible for the review and evaluation of plant operational experience and of off-normal events.

For the permanent situation, degreed engineers will be provided with additional training and assigned on shift as STAs. This group of engineers will also perform the operational analysis function.

We conclude that the licensee has satisfied the Category "A" requirements for the shift technical advisor. Verification of the licensee's procedures for implementation of this item will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

The licensee has revised plant procedures to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift.

Based on the above considerations, we conclude that the licensee has satisfied the requirements of Item 2.2.1.c to provide new procedures. Verification of the implemented procedures will be performed by the Office of Inspection and Enforcement and will be documented by an appropriate inspection report.

2.2.2.a CONTROL ROOM ACCESS

The licensee has implemented procedures which will limit control room access during an emergency. These procedures establish the authority and responsibility of the person in charge of the control room to control access and establish the line of succession for the person in charge of the control room. We therefore conclude that the licensee has satisfied our requirements of this item. Verification of licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.b ON-SITE TECHNICAL SUPPORT CENTER (TSC)

The licensee has established an onsite technical support center on the 4th floor of the Plant Administrative Building which is located south of the turbine building. Direct telephone communication between TSC and the control room, near site Emergency Operations Center and the NRC has been established. Portable airborne and radiation monitors have been provided for the TSC to provide warning and monitoring capability. Access to technical data (plant drawings and records) is available from the plant's records management system. This is located within one floor of the TSC. Plant parameters can be monitored by CRT display in the TSC from the plant computer.

For the long term Category "B" requirements, the licensee has proposed two Technical Support Centers. A Primary Technical Support Center (PTSC) and a Secondary Technical Support Center (STSC). The PTSC will be located in the administrative building as discussed above. The STSC will be located approximately .65 miles from the plant. It should be noted here, however, that PTSC is not habitable because of shielding problems in the administrative building, while the STSC will be habitable when completed. This proposal is under NRC consideration and will be reviewed as a Category "B" item.

Based on our review of licensee's submittal and our site visit, we have concluded that the TSC at the ANO-2 satisfies our short term requirements set forth in NUREG-0578. Verification of the licensee's procedures for the short term TSC will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.c OPERATIONAL SUPPORT CENTER

The licensee has established an on-site Operations Support Center (OSC) located in the vicinity of the on-site technical support center. Operations Support Personnel will be located in the OSC for response to the control room and/or TSC needs. Telephone communications and the station page system presently exist with the control room and the TSC.

Based on our review of licensee's submittal and our site visit, we conclude that the licensee has satisfied our requirements set forth in NUREG-0578. Verification of licensee's procedures to cover this center will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

NRR REACTOR COOLANT SYSTEM HIGH POINT VENTS

The licensee has proposed installation of remotely operated vents for the reactor vessel head and for the top of the pressurizer. Each vent point will have parallel redundant paths with two valves in series. All vent paths will be safety grade. The proposed design satisfied the short-term requirements and is acceptable. Implementation of the proposed design is a Category "B" item.

REFERENCES

1. Letter, APLC (Cavanaugh) to NRC (E/NRR) October 17, 1979.
2. Letter, APLC (Cavanaugh) to NRC (E/NRR) November 20, 1979.
3. Letter, APLC (Cavanaugh) to NRC (E/NRR) December 5, 1979.
4. Letter, APLC (Cavanaugh) to NRC (E/NRR) December 17, 1979.
5. Letter, APLC (Cavanaugh) to NRC (E/NRR) January 17, 1980.
6. Letter, APLC (Trimble) to NRC (E/NRR) January 18, 1980.
7. Letter, APLC (Trimble) to NRC (E/NRR) January 29, 1980.
8. Letter, APLC (Trimble) to NRC (E/NRR) January 31, 1980.
9. Letter, APLC (Trimble) to NRC (E/NRR) February 29, 1980.
10. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, dtd. September 13, 1979.
11. Letter, NRC (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, dtd. October 30, 1979.