October 1, 1968

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ARKANSAS POWER AND LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT

DOCKET NO. 50-313

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1.0 INTRODUCTION

On November 29, 1967, the Arkansas Power & Light Company (applicant) submitted an application to construct and operate a single-unit nuclear power plant, to be known as the Russellville Nuclear Unit. Ten supplements to that application have since been filed with the Atomic Energy Commission. The reactor site is located about 6 miles from Russellville on a peninsula in the Dardanelle Reservoir on the Arkansas River, Pope County, Arkansas.

The facility architect-engineer and construction manager will be the Bechtel Corporation, the nuclear steam supply system will be furnished by the Babcock & Wilcox Company (B&W), and the turbine generator will be supplied by the Westinghouse Corporation.

The plant will use a B&W pressurized water reactor designed to operate at 2452 megawatts thermal (Mwt) and produce 850 megawatts of electrical power (Mwe). The expected ultimate capacity of this plant is 2568 Mwt. The applicant has designed the major plant components including the containment and other engineered safety features for a power level of 2568 Mwt, and has used this power level in analyzing postulated accidents in conformance with the siting guidelines of Title 10 - Chapter I, Part 100 of the Code of Federal Regulations (10 CFR 100). We evaluated the containment and other engineered safety features for 2568 Mwt; however we evaluated the thermal and hydraulic characteristics at 2452 Mwt. Before operation above a power level of 2452 Mwt is authorized, the Commission's regulatory staff must perform a safety evaluation to assure that the facility can be operated safely at that power level.

The application, including the Preliminary Safety Analysis Report (PSAR) and Supplements 1-10 (hereinafter collectively referred to as the "application") was the basis on which the Division of Reactor Licensing conducted the technical evaluation of the preliminary design of the proposed plant. The staff used the following approach in its review of this application.

a. Performed an in-depth evaluation of site-related features.

b. Identified and compared all of the design and safety features of the Russellville Nuclear Unit for similarity to those previously reviewed. Where justified, we relied upon previous in-depth evaluations of like systems, components, and structures without performing separate, duplicate evaluations.

c. Determined that the design features and the treatment of safety matters were consistent with current regulatory criteria and policy, and that the applicant adequately addressed concerns which have been identified by the Advisory Committee on Reactor Safeguards (ACRS) in previous reviews.

d. Identified and evaluated those design features and related safety matters that are new or unique, or which, although reviewed in the past for other applications, continue to require review.

Within the Division of Reactor Licensing, the Reactor Projects group was responsible for the review, and for coordinating parts of the review involving personnel within the Division representing various special technical disciplines from the Reactor Technology and Reactor Operations groups, as well as consultants and other governmental agencies outside of the Division of Reactor Licensing. The reports of our consultants are attached as Appendices C through G.

During the review a number of meetings were held with representatives of the applicant to discuss the proposed plant. As a consequence, additional information was received from the applicant.

The Advisory Committee on Reactor Safeguards has considered the application, has visited the site, and has met with both the applicant and the staff. A copy of the ACRS report to the Commission on the Russellville Nuclear Unit is included as Appendix A.

A chronology of the principal actions relating to the processing of the application is attached as Appendix B to this report.

The review and evaluation of the proposed design and construction plans of the applicant prior to construction constitute the first stage of a continuing AEC review of the proposed facility. Prior to issuance of an operating license, the Commission's regulatory staff will review the final, as-built, design and operating features to determine that all of the Commission's safety requirements have been met. The unit would then be operated only in accordance with the terms of the operating license and the Commission's regulations, and under the continued surveillance of the Commission's regulatory staff.

The issues to be considered, and on which findings must be made by an Atomic Safety and Licensing Board before the requested construction permit may be issued, are set forth in the Notice of Hearing published in Federal Register on September 20, 1968, 33 FR 14243.

2.0 SITE AND PLANT DESCRIPTION

2.1 Site Description

The Russellville Nuclear Unit will be constructed on an 1100 acre site located on a peninsula in the Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas approximately 6 miles from the town of Russellville (1967 population, 11,154) and 2 miles from the village of London (1967 population, 495).

An exclusion area with a radius of 0.65 miles (3430 feet) from the reactor has been established for this plant. All land within this radius, except for the bed and banks of the Dardanelle Reservoir is owned by the applicant. The bed and banks of the reservoir are controlled by the U.S. Army Corps of Engineers. The applicant has obtained an easement from the Corps of Engineers for the area which will permit it to exclude all persons from this area in the event conditions at the plant warrant such action. The applicant has specified a low population zone (LPZ), as defined in 10 CFR 100, of 4 miles.

The area around the site is largely undeveloped. In 1964 practically no land was under cultivation out to 4 miles; out to 10 miles less than 0.4 percent was under cultivation. In 1964 approximately 20% of the land out to 5 miles and 27% of the land out to 10 miles from the site was classed as pasture land. The nearest population center with over 25,000 people is Hot Springs, Arkansas, 55 miles south of the site. The applicant has estimated a 1967 population of 3146 within 4 miles (LPZ) and 22,993 within 10 miles of the site. Projections of the total population within these distances have been made by the applicant for the year 2012 and are given as 5700 and 34,827, respectively.

The meteorology of the site is typical of continental locations, with lighter wind speeds and slower diffusion conditions at night than during the day. The site is in an area with appreciable tornado activity with 41 tornadoes reported per 1 degree square[1/] over a 45-year period (1916-1961).

With respect to hydrology, the maximum probable flood, as computed by the Corps of Engineers, combined with failure of the upstream dam will flood the reactor site to 361 feet or 8 feet above plant grade level. An onsite pond which will provide the source of emergency cooling water will be available in the unlikely event that there is a loss of such cooling water from the Dardanelle Reservoir.

In terms of geology, the site is near the axis of the Scranton syncline, one of several westward-trending gentle folds that characterize the Arkoma Basin--a major structural and topographic feature of Arkansas and eastern Oklahoma. The site is underlain by a thick sequence of gently-dipping shales and sandstones of Pennsylvanian age. Overburden consists of alluvial clay and silty clay that ranges in thickness from 13 to 23 feet.

No identifiable active faults or other recent geologic structures exist that would localize earthquakes in the immediate vicinity of the site. Although several ancient faults are associated with the folded structures in the area, none appear to have been tectonically active since latest Paleozoic time (about 230 million years ago).

1/ A 1 degree square as used here is that earth surface area bounded by 1 degree of lattitude and 1 degree of longitude. At the Russellville site a 1 degree square contains approximately 3000 square miles.

A somewhat unique site feature is the buried natural gas transmission pipe line which crosses the site approximately 600 feet from the containment structure. The line, which does not supply this facility, will cross 4 feet beneath the bed of the plant's discharge water canal.

A discussion of the acceptability of the site is given in Section 3.1.

2.2 Plant Description

The Russellville plant will have a closed-cycle, pressurized-water nuclear steam system housed in a prestressed concrete containment building, a steam and power conversion system housed in an auxiliary building and an outside electrical switchyard. It will also have those auxiliary systems and structures required to safely operate and maintain the plant under normal and emergency conditions. These auxiliaries include a radioactive waste disposal system, fuel storage and handling facilities, emergency power systems, and other engineered safety features.

The principal features and design bases for the Russellville Nuclear steam supply system are essentially identical to those of the Metropolitan Edison Company's Three Mile Island Nuclear Station, for which a construction permit has been issued by the Commission. The nuclear steam supply system consists of a pressurized water reactor, a reactor coolant system, and associated auxiliaries. The reactor coolant system consists of two parallel recirculation circuits, each sending reactor coolant through a steam generator (reactor coolant side) where it splits and flows through two pumps and associated piping, back to the reactor vessel.

An electrically-heated, spray-cooled pressurizer is connected to one of the two flow circuits. The reactor core uses fuel rods of uranium dioxide pellets clad in Zircaloy-4 tubes. The fuel rods are supported in assemblies by spacing grids and fittings, and a perforated can, all made of 304 stainless steel. Reactivity is controlled by movement of control rods (Ag-In-Cd), clad with 304 stainless steel, and by varying the boric acid concentration in the reactor coolant.

The control rods are positioned axially in the core by the use of electro-mechanical, rack-and-pinion rod drive mechanisms and tripped (gravity insertion for least reactivity) by deenergizing a magnetic clutch. The clutch design permits the drive motor to apply down-drive force should a rod not fall freely.

A control system monitors reactor system temperatures, pressure, flows, neutron flux and load demand, and adjusts reactor power, steam generator feedwater flow, and turbine throttle within prescribed operating limits.

A reactor protection system monitors reactor coolant system temperatures, flows, and pressure, core neutron flux startup rate, and neutron flux level. If an operating limit is reached, this system shuts down the reactor by releasing rod drive clutches and allowing the control rods to drop into the core.

The principal engineered safety features are the emergency core cooling system (ECCS), the containment ventilation system, and the containment spray system (with chemical additives). A protection system monitors primary coolant and reactor building pressures and will automatically initiate operation of the engineered safety feature systems if preestablished safety limits are reached.

The containment structure will be a steel-lined, prestressed, post-tensioned concrete, vertical cylinder with flat bottom and shallow domed roof. The containment is of the same basic design as that used by Bechtel for the Commission-licensed Duke Power Company Oconee Units 1, 2, and 3, the Florida Power & Light Company Turkey Point Units 3 and 4, and the Consumers Power Company Palisades Plant. The design details of the Russellville containment differ from this basic design in that the design details provide for a modified prestressing system using three vertical buttresses and 240°-span horizontal tendons rather than for a prestressing system using six vertical buttresses and 120°-span horizontal tendons.

All penetrations will be pressure-resistant, leak-tight, welded assemblies. Personnel hatch openings will have interlocked double doors, the equipment hatch will have a double-gasketed, bolted door and an isolation system will be provided to close all fluid lines that penetrate containment and are not required for operation of engineered safety features.

The emergency core cooling system (ECCS) will be designed to provide core cooling for any location and size primary coolant pipe break, up to and including the double-ended rupture of the largest pipe--the 36-inch reactor outlet pipe between the reactor pressure vessel and the steam generator. The ECCS will consist of two operating and one spare high pressure injection pumps, two core flooding tanks (accumulators) and two low pressure (decay heat) pumps. A recirculation system using the two low pressure pumps, returns water from the containment sump to the ECCS. The Russellville ECCS does not differ in concept or capacity from the ECCS reviewed and approved for Metropolitan Edison's Three Mile Station Unit 1.

An emergency containment spray system will provide borated water containing dissolved sodium thiosulfate and sodium hydroxide to limit containment accident pressure (by heat removal) and to remove iodine (by chemical action) in the event of an accidental energy release from the primary system. A containment ventilation system, consisting of three fin-fan air coolers, is used to maintain containment temperatures at normal values during normal plant operations. During accident conditions either the coolers alone or the core spray system alone will be capable of keeping the accident pressure within the design limit.

The major plant auxiliary systems are the chemical and volume control system, the waste disposal system and the fuel handling system. The chemical and volume control system is used to adjust the concentration of the chemical neutron absorber (boric acid) in the reactor coolant and to maintain the proper amount of water in the primary system. The waste disposal system is used to accumulate radioactive gases, liquids and solids from plant operation, process the radioactive wastes, and control and monitor the release of radioactive gases and liquids from the plant to the air and to the reservoir respectively. The fuel handling system includes equipment and facilities designed to transport spent fuel under water from the reactor to the water-filled spent-fuel storage pool from where the spent fuel will be shipped to an offsite processing plant.

The closed steam-feedwater cycle of the steam and power conversion system removes heat energy from the reactor coolant in the two once-through steam generators in the form of steam, converts steam energy into electrical energy in passing through the turbine generator, condenses the steam into feedwater which is purified, chemically controlled for optimum pH and minimum oxygen content, preheated, and recycled to the steam generators.

The condenser circulating water system condenses the steam leaving the turbine generator unit in the main condenser. The pumps for this system will withdraw water from the Dardanelle Reservoir by way of an intake canal and pump it through submerged conduits to the main condenser, and thence back to the reservoir through submerged conduits and a discharge canal. Cooling water for vital plant functions, which must remain operable in the event of an accident, will be supplied by the service water system. This system will draw water from an intake structure which is normally supplied through the intake canal from the Dardanelle Reservoir. The service water portion of the intake structure can be isolated from the intake canal and be gravity-fed by submerged piping from an elevated emergency cooling water pond to be constructed on the site.

Onsite emergency power to operate post-accident emergency core cooling systems, the containment cooling systems, and other vital systems will be supplied by two 2750 kW diesel generators. Two separate 125 volt d.c. systems, complete with charged storage batteries, will also be provided to supply vital instrumentation and provide emergency lighting and switching power.

There are two independent offsite sources of power. Offsite power can be provided automatically upon loss of the main generator, through one of two transformers from a 161 kV transmission system which will be supplied power over separate lines from different sources. Offsite power can also be provided automatically in a similar fashion from a 500 kV transmission system which can also bring power into the plant over separate lines from two different sources.

3.0 IMPORTANT SAFETY CONSIDERATIONS

In our evaluation of this application, we have given special consideration to a number of site and design features which are new, unique, require continuing evaluation, or have important safety implications. The more important of these safety considerations are discussed in the following sections.

3.1 Suitability of the Site

In evaluating this reactor site, we have considered the following aspects: the characteristics of the proposed reactor; the containment capability; the nature and amount of radioactive waste products generated; the site characteristics relating to meteorology, hydrology, geology, and seismology; abnormal weather conditions, such as tornadoes and floods; the population distribution in the surrounding area; and the potential radiation exposures at the site boundary and offsite as a consequence of any of the postulated design basis accidents.

The area around the site is sparsely populated; however, the site does present one potential problem related to evacuation of the few persons on the Bunker Hill section of the peninsula which extends into the Dardanelle Reservoir. Since the land evacuation route for these people would be across the applicant's property inside the exclusion area, the applicant will provide boats to evacuate these persons by water if the land route is unsafe.

The applicant states that no water is removed for either industrial or potable purposes downstream between the plant and the Mississippi River.

To establish background radiation levels, the applicant has outlined an environmental program which will be initiated 12 to 18 months prior to operation of the Russellville plant. This program will include onsite monitoring of radiation exposure levels and radionuclide concentrations in soil, vegetation, lake bottom, water, fish, and air. Offsite monitoring will include analyses of milk, pasture forage, truck crops, and public water supplies. The applicant has consulted with various state and federal agencies in establishing this program.

The applicant's program has been reviewed by the Fish & Wildlife Service (Appendices C1 and C2). The Fish & Wildlife Service has recommended that the applicant's program include pre- and post-operational survey studies regarding specific radionuclides and their effect on selected organisms indigenous to the area. On the basis of our review of supplementary information submitted in response to Question 2.9 in Supplement No. 3 of the PSAR, we conclude that the applicant intends to comply with these recommendations of the Fish & Wildlife Service.

We conclude that with the incorporation of these recommendations, the applicant's proposed program is acceptable.

On the basis of available data, we conclude that the site meteorology does not present any unusual problems. However, to supplement and verify the existing data, the applicant has indicated that an onsite meteorological measurement program will be conducted. We find the scope of this program to be acceptable. The applicant's meteorological assumptions relating to site diffusion factors are considered to be adequately conservative. This finding is based on independent analyses performed by the staff and by the Environmental Science Services Administration, whose comments are attached as Appendix D.

To meet our safety criteria, certain aspects of the site required further definition and/or changes to the material originally presented in the application. These matters are discussed in the following paragraphs.

The applicant added an emergency cooling water pond on the site to ensure, in the unlikely event of failure of the Dardanelle Lock and Dam, a continued source of emergency cooling water for vital plant functions.

The onsite gas transmission line, described in Section 2.1, has been evaluated for effects on the Russellville plant. The buried line, at its nearest point, is 600 feet from the reactor containment building and is 4 feet below the bed of the discharge canal. The applicant has calculated the energy potential for this line due to an explosive rupture and that due to ignition of gas discharged from the open line. Neither the explosive rupture nor the radiant energy from gas ignition at the break are considered capable of damaging this facility. The applicant has further indicated that, should such events occur, the gas line owner will close control valves on both sides of any break in the plant vicinity within 2 hours of notification. The applicant has stated that prior to plant operation, the existing pipe will be replaced with pipe constructed to Type C specification of ASA Code B 31.8 for a distance of 600 feet on each side of the crossing. On the basis of our review of this information and analysis, we do not consider this pipe line to be a significant hazard to the safe operation of this plant.

The geology of the site was found to be generally favorable by us and the U.S. Geological Survey, whose report is attached as Appendix E. In summary, our review shows that the site is underlain by shales and sandstones of Pennsylvanian age. Overburden consists of alluvial clay and silty clay that ranges in thickness from 13 to 23 feet. No identifiable active faults or other recent geologic structures exist that would localize earthquake in the immediate vicinity of the site. The limited subsurface data available indicate that the major units of the nuclear facility will be founded on a hard, dense shale which should provide an adequate foundation.

Considering the site geology, soil conditions and earthquake history, the U.S. Coast & Geodetic Survey (USC&GS) and we concluded that an acceleration of 0.1 g would adequately represent earthquake disturbances likely to occur within the lifetime of the facility and that an acceleration of 0.2 g would adequately represent the ground motion from the maximum earthquake likely to affect the site. The applicant will use these parameters in the seismic design of all Class I structures and systems. The USC&GS report is attached as Appendix F.

The applicant's original design criteria considered tornadoes having a tangential velocity of 300 mph, translational wind velocity of 40 mph, and a barometric pressure drop of 3 psi in 5 seconds. Following discussion with the regulatory staff, the applicant agreed to change these criteria to design for a tornado having a translational wind velocity of 60 mph and a barometric pressure drop of 3 psi in 3 seconds. Design basis missiles equivalent to a 4-inch by 12-foot plank traveling end-on with a velocity of 300 mph at any height and a 4000-lb auto traveling through the air with a velocity of 50 mph at a height of 25 feet or less are proposed. These values are consistent with values used by other nuclear plants recently approved for construction in areas having a significant history of tornado activity and, in our judgment, are reasonable design criteria. We conclude that the tornado design bases including the effects of tornado-generated missiles are acceptable.

In the unlikely occurrence of the maximum probable flood concurrent with the failure of the upstream Ozark Dam, the site would be flooded to a level of 361 feet which is 8 feet above plant site grade level. The applicant has considered this in the facility design and has stated that all vital equipment including service water cooling pumps either will be located above maximum probable flood level or will be protected by waterproof Class I structures. We therefore conclude that the applicant will provide adequate flood protection for this facility.

We conclude that the applicant has adequately considered the important characteristics of the proposed site. We find the proposed site to be acceptable.

3.2 <u>Acceptability of the Nuclear Steam Supply System Design</u>

The reactor design characteristics for the Russellville Plant are essentially the same as those for the Commission-approved Three Mile Island, Crystal River, and Rancho Seco plants. As in those plants, operation will be at 2452 Mw thermal with a maximum fuel burnup of 5[illegible] megawatt-days per metric ton of uranium (Mwd/MTU).

During part of the first fuel cycle the core is predicted to have a slightly positive moderator temperature coefficient of reactivity. Present calculations indicate that, with this coefficient, the core could withstand a loss-of-coolant accident and not exceed 2000° F peak fuel clad temperature. An acceptable final design value of the positive moderator temperature coefficient will be set at the operating license stage. The applicant has agreed to reduce or eliminate this positive coefficient, if necessary, to bring the consequences of the applicable accident within acceptable limits.

B&W has provided for the evaluation of xenon oscillations and in-core neutron detectors in its research and development program. To date, calculations have been performed which indicate that xenon oscillations are not expected in the azimuthal or radial direction, and are not likely in the axial direction at any time during the initial fuel cycle. Further analyses will be made using final values of core properties. Calculations have also been made to show feasibility of controlling a divergent xenon oscillation using part-length control rods. Since xenon oscillations are relatively slow flux variations which could be detected by the proposed in-core flux instrumentation, we believe that such a control technique is feasible and could be provided.

The above-mentioned in-core flux instrumentation consists of 52 fixed-position self-powered flux detectors distributed throughout the core. Normal readout is provided by the plant computer. Data obtained from this system will provide a history of fuel burnup, power distribution, and power disturbances during operation. In the event that the plant computer fails, there is an alternate readout system for selected in-core detectors.

With respect to the thermal-hydraulic parameters and design features, our review revealed nothing new or different from recently authorized pressurized water reactors. However, as noted in Section 4.0, additional analytical and experimental verification to support the choice of the fuel damage limit, the use of stainless steel shims and the use of part-length rods will be obtained before the Russellville plant receives an operating license.

We have reviewed the applicant's seismic design bases pertaining to the reactor vessel, reactor internals, and other Class I (seismic) mechanical systems and components. These systems will be designed to withstand normal design loads of mechanical, hydraulic, and thermal origin, plus applicable earthquake loads, as well as concurrent accident-induced blowdown loads. Our evaluation of the proposed design criteria for reactor internals and Class I mechanical systems and components indicates that they will provide an adequate margin of safety.

One aspect which we are reviewing in detail is that of thermally-induced stresses in the pressure vessel during actuation of the emergency core cooling system. The initial results of the applicant's analysis of this accident indicate that no loss of vessel integrity would be experienced even if large flaws were presumed to exist in the vessel wall at the beginning of the quenching. However, in view of the uncertainties associated with the analytical methods used to arrive at these results, the applicant plans to continue his work on this problem. While there remain uncertainties in the analyses being pursued, it is important to note that there is a significant time available (about 5 years) until material properties will be affected by irradiation to an extent that will be of concern. Further, it appears that there are means that can be employed, if necessary, to reduce the potential for vessel failure resulting from thermal shock and to mitigate the consequences of such a failure should it occur.

As recommended by the ACRS (Section 5.0), we will continue to review information subsequently developed concerning thermal shock on the pressure vessel to ensure that the calculational models used are not in conflict with experimental data.

Provided that the development program substantiates the reactor design characteristics discussed above, we conclude that the design of the nuclear steam supply system is acceptable.

3.3 Engineered Safety Features Adequacy

Engineered safety features for this plant include the emergency core cooling system (with reactor vessel internal vent valves), the containment ventilation systems, and the containment spray systems, and associated iodine removal system.

The emergency core cooling system (ECCS) is described in Section 2.2 of this report. The applicant's design basis is the same as that of Crystal River and other Babcock & Wilcox-designed systems recently reviewed. That basis is to prevent fuel clad melting for the entire spectrum of reactor coolant system failures from the smallest leak to complete severance of the largest reactor coolant pipe. To provide assurance that this criterion is met and to prevent any mechanical damage that might interfere with core cooling, the applicant has sized the emergency core cooling systems to limit the clad temperature transient to 2300° F or less. The calculated peak clad temperature, about 1950° F, occurs transiently during the postulated hot-leg break (a 14.1 ft^2 break).

We have reviewed the applicant's failure mode analysis of the ECCS and have concluded that adequate short-term cooling can be provided at high and low vessel pressures even in the event of failure of any single active component. In addition, adequate redundancy is provided to accommodate failure of a single active or passive component without jeopardizing the ability for long-term core cooling with the ECCS in the recirculation mode. To achieve this, the applicant revised his originally proposed ECCS design to provide two systems with no sharing of active components and minimum practical sharing of passive components. This applicant's ECCS as revised is the same as those systems previously reviewed. The result is that there are two separable core cooling systems which share only the passive borated water storage tank, core flooding tanks, and containment building sump. Sharing of the tanks is acceptable since they are in use for only a short period of time. Sharing of the reactor building sump is acceptable since the recirculation lines for the two systems take suction from different locations of the sump, the sump is covered with a grating and heavy duty strainers are provided.

As was done in the B&W nuclear steam supply system design provided for the Three Mile Island, Crystal River, and Rancho Seco plants, the Russellville design incorporates one-way internal vent valves in the reactor core barrel to prevent steam binding above the core. In the event of a loss-of-coolant accident initiated by a break in a cold leg of a reactor loop, the valves will open to permit steam generated in the core to flow directly to the leak and thus not prevent the emergency core coolant system from keeping the core adequately covered. These valves have been previously authorized for use in the Three Mile Island plant.

B&W has made a preliminary sensitivity analysis using worst case parameters to show how loss of core flow, by shunting reactor coolant through a failed (open) valve, affects the DNB ratio (design limit is 1.3 or greater). The preliminary analysis shows the reduced-flow DNB ratio is 1.68 at 100% power, 1.30 at 112% power and 1.24 at 114% power (the highest thermal power calculated in any operational transient). An analysis based on the final design of the core is expected to meet the 1.3 DNB ratio design requirement at 114% of rated power as well. We also considered the ability to detect, by change in measured reactor coolant loop flow, the failure of more than one vent valve. Based on the preliminary design data supplied by B&W, the total system flow is increased by 1.1% by failure of one valve. The applicant has stated that flow distribution studies will be made using a model of the reactor to simulate failure of the vent valves. Completion of valve testing, including vibration tests, is expected by January, 1969. At the operating license review on this plant, or earlier, we will evaluate the results of these tests and verify the ability to identify failure of the vent valves by detection of changes in reactor coolant flow. We conclude, at this stage of our review, that the vent valve design is satisfactory subject to completion of the final design, design analyses, testing, and verification of ability to use flow change to detect failure.

Two diverse methods are provided for containment heat removal under accident conditions: (1) two 120 x 10^6 Btu/hr capacity containment spray systems, each of which takes relatively cool water (initially from the borated water storage tank and later from the containment sump) and delivers it to the containment atmosphere through a spray header and (2) three 80 x 10^6 Btu/hr capacity containment cooling systems, each consisting of a fan and tube cooler, which removes heat from the containment air and transfers it to the low-pressure service water system.

The containment cooling requirement is that the post-blowdown reactor building pressure be maintained below the containment design pressure. This requires an initial heat removal capacity of about 240 x 10^6 Btu/hr. This requirement can be satisfied if either all sprays or all containment cooling systems are assumed to be inoperative. It can also be satisfied if one spray and one cooler are inoperative. On the basis of our review of these systems, we conclude that adequate capacity has been provided to initially limit and subsequently reduce the containment pressure (and thereby reduce leakage) after the design basis accident, in the event such an improbable accident should occur.

A chemical additive (sodium thiosulfate with sodium hydroxide) will be mixed with the spray water to remove iodine from the containment atmosphere following a loss-of-coolant accident. Two spray systems are provided as discussed above. Each spray system has the design capability to deliver an adequate amount of the chemically treated spray to the containment atmosphere to prevent exceeding 10 CFR 100 guidelines for potential radiological doses at the site boundary and at the low population zone boundary. Section 3.7 gives the calculated doses using a single spray system and also states that, in the event additional chemical iodine spray tests now underway indicate that the spray system is not as effective as anticipated, iodine reducing charcoal adsorber units can be added to remove iodine.

The service water system shown on Figure 9-4 of the PSAR provides all water required for emergency cooling of the engineered safety feature equipment including the containment building coolers and the emergency diesel generators. Redundant pumps and piping and an emergency reservoir are provided such that no single failure can cause loss of required cooling.

3.4 Foundation and Structural Design Adequacy

In evaluating the foundation and structural design of the plant structures, we and our consultant,[1] considered the following general aspects: the geology and nature of the subsoils, the seismic design parameters, site flooding, tornado wind loadings, and the effects of missiles generated from tornadoes and internal plant sources. We considered the following specific aspects in our evaluation of the containment and other Class I structures: design criteria, specifications and inspection for concrete reinforcing, selection of loads, load combinations and allowable stresses for the structure, liner and liner anchorage criteria, tendon and tendon anchorage criteria, design of penetrations, and containment strength and leak testing.

All structures and equipment required for plant safety and to maintain the integrity of engineered safety feature systems have been designated as Class I. All other structures and equipment are Class II. All Class I structures will be designed to behave elastically under normal and accident loads, except that limited yielding will be permitted under a combination of dead load, piping thermal shock or rupture, and design-basis earthquake (0.2 g). Class II structures, which do not perform vital safety functions, will be designed to Zone 1 requirements of the Uniform Building Code. Class II equipment will be designed for an equivalent horizontal loading of 0.05 g.

1/ Nathan M. Newmark Consulting Engineering Services. See report attached as Appendix G.

The containment building, as noted earlier, is similar to other Bechtel designs including the design for the Rancho Seco plant, except that heavier tendons and three instead of six buttresses are used. The cylinder has staggered 240°-span instead of 120°-span horizontal tendons. The vertical and the dome tendon systems are similar to those used in previous designs, except for anchorage designs and tendon sizes.

In response to our questions on several aspects of structural design, the applicant provided additional supporting details on methods of analysis, and construction details.

We and our consultants have reviewed the proposed tendon systems tentatively selected by the applicant. We conclude that use of the tendon systems proposed, with up to 184 wires per tendon, would be acceptable.

The liner anchorage design is similar to that proposed for the Rancho Seco plant. The liner anchorages are designed to fail before the liner itself can fail. We have expressed concern that, with the liner in compression and tending to buckle locally, anchors may fail rapidly and sequentially. On the basis of our review, we do not believe the analyses presented in the PSAR are conclusive. We have discussed this with the applicant and (as noted in Section 4) prior to construction we will obtain confirmatory test specimen data that deal with gross liner failure considerations.

In the tendon anchor zone, we are concerned that sufficient reinforcing be included in the design to cover all possible tension stresses that may exist in this zone. The usual design methods neglect two potentially significant tensile stresses, those generated by temperature gradients and by concrete shrinkage. As noted in Section 4, prior to construction of these anchorages, we will obtain from the applicant analyses and qualification test data to confirm design adequacy.

The problem of tornado-induced loss of water from the fuel pool leading to fuel melting and fission product release is of continuing concern We have examined the analysis provided by the applicant in this regard and find that it contains no new information or arguments that have not been presented in previous applications.

We are continuing to examine the requirements for spent fuel pool design and we conclude that the design of the fuel storage pool should be such that protection of the pool from water removal effects could be added if this is found necessary. The applicant has agreed to provide this capability in the design of the Russellville fuel storage pool.

The applicant has proposed a 2% statistical sampling program for strength testing of the Cadweld reinforcing bar splices to be made in the structures. Since this may result in a small number of welds being tested, we are examining the area further. In the event that a modified testing program is considered necessary requiring a larger number of welds being tested or placing more emphasis on selected weld locations, we conclude that these relatively minor changes can be agreed upon with the applicant prior to the actual placing of these welded splices in the structures.

From our in-depth review, we and our seismic design consultant conclude that the containment, foundation and general structural designs proposed for the Russellville plant are acceptable except for the submission of confirmatory data on liner and tendon anchorages and Cadweld splice tests. These items have been left for later consideration as discussed in Section 4.0.

3.5 Adequacy of Instrumentation, Control and Emergency Power Systems

The instrumentation and control systems were evaluated and found to comply with the Commission's General Design Criteria (see Section 3.8) and IEEE 279, Proposed Criteria for Nuclear Power Plant Protection Systems. A comparison was also made with the systems proposed for the Three Mile Island Nuclear Station, and Crystal River Unit No. 3. The applicant has verified, and we concur, that the proposed design of the instrumentation and control systems for the Russellville plant and the above mentioned plants are substantially identical in concept except that the Russellville plant (a) uses not one, but all four of the redundant reactor power level channels in an averaging system as inputs to reactivity control; (b) initiates reactor trip upon loss of any two pumps while the other plants utilize systems which permit continued operation with the loss of one pump in each loop provided power is below a predetermined safe limit; (c) supplements reactor coolant systems code safety valves with a pilot actuated relief valve which is not provided in the other plants; and (d) varies boiler feed pump speed as the major means of controlling feedwater flow as opposed to reliance in the other plants solely upon feedwater valve control. The differences noted in (b), (c) and (d) above are considered to be minor and to have no significant effect on reactor safety. In evaluating item (a) we examined the proposed design and found it to be in compliance with IEEE 279. In particular, the protection system has four redundant power level channels. The random failure of any one channel leaves three for protection, only two of which are required. While these channels are also connected to the plant's reactivity control system, a single random failure in any one channel is prevented from causing a control failure by isolation devices and by the manner in which they are combined. Further the applicant reports that tests have been successfully performed simulating open circuits, short circuits, grounds, and faults to high voltages with no failures propagating beyond the channel in which the simulated failure was imposed.

As a result of our evaluation of item (a) we conclude that the design provides satisfactory protection against random failures. We will continue to work with the applicant to ensure that it takes into account, in completing the design of protection and control instrumentation, the possibilities of common failure modes such that by the suitable use of redundant devices with functional and equipment diversity, the proposed interconnections of protection and control instrumentation will not adversely affect plant safety.

The control room contains instrumentation and controls necessary for safe operation of the nuclear facility. Safe occupancy of the control room during abnormal conditions is provided for in the design. In the event the control room becomes uninhabitable, sufficient instrumentation and controls are provided at local stations which permit the operator to maintain the reactor in a hot standby condition. Further, the applicant has stated that the capability to perform an orderly cold shutdown from outside the control room, should this room become inaccessible for a long period of time, will be provided. We conclude that the control room design bases meet the intent of Criterion 11 of the General Design Criteria.

The applicant has established criteria for the selection, protection, and routing of all control, power and instrumentation cables. We conclude that adequate measures will be taken to prevent and minimize the possibility of fire or other damage in electrical cabling.

We have evaluated the proposed offsite and onsite electric power systems and have concluded that they comply with Criterion 39 of the General Design Criteria.

In its letter on the Three Mile Island Nuclear Station, the ACRS recommended that consideration be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element. The applicant has indicated that it will provide continuous radiation monitors in the reactor coolant makeup and letdown line and in the containment atmosphere sample line with sufficient sensitivity to promptly detect a gross fuel element failure. Information on the response time as a function of fuel failure severity will be made available during the detailed design of the plant. We will review this matter on other plants scheduled for operation before the Russellville plant, and at the operating license stage review of the Russellville plant.

On the basis of the foregoing, we have concluded that the reactor instrumentation, control, and emergency power systems are acceptable for this construction permit stage of review.

3.6 Radioactive Waste Disposal Adequacy

The radioactive liquid wastes generated in normal plant operations will be collected, stored, treated, measured for activity, and discharged on a batch basis with continuous moni oring during discharge through a line to the plant's circulating water discharge canal. Gaseous wastes will be collected, monitored, diluted and released to the atmosphere. If the activity levels exceed precribed limits, the gases will be compressed and stored in waste gas decay tanks. Following decay, the stored gases will again be monitored prior to release to assure that release is within prescribed limits. Solid radioactive wastes accumulated from plant operation will be temporarily stored onsite. Shipment from the site will be in containers approved for that purpose.

We reviewed the possibility of activity release due to system failures. The solid and liquid disposal equipment is located in shielded, controlled-access areas of a Class I structure with provision for contamination control in the event of spills or leakage. Calculations by us and the applicant indicate that failure of a waste gas tank containing maximum activity would result in whole body doses of less than 2 rem at the site boundary which is well below 10 CFR 100 limits.

On the basis of our review, we conclude that the proposed radioactive waste disposal system will adequately control the radioactive wastes generated from plant operations.

3.7 Analysis of Radiological Consequences from Potential Accidents

Potential accidents which could result in radioactive releases to the environment have been analyzed by the applicant. We have evaluated these accidents and the engineered safety features provided to mitigate or limit the potential offsite exposures. Accidents which have been considered are: the loss-of-coolant accident, the rod-ejection accident, rupture of a steam pipe, rupture of a steam generator tube with loss of offsite power, fuel-handling accident, accidental release of radioactive liquid and gaseous waste, and rupture of a recirculation line in the emergency core cooling system. Of those accidents considered to have a potential for significant releases of radioactivity to the environment, the loss-of-coolant accident would result in the highest potential offsite doses.

For accidents involving loss of coolant from the primary system, the emergency core cooling systems are designed to limit fuel cladding temperatures to well below the melting temperature, to prevent shatter of the fuel cladding, and to limit fission product release from the fuel. However, for conservatism we assume that the containment and its associated engineering safety features must be capable of limiting potential doses in conformance with 10 CFR Part 100 guidelines assuming releases of fission products from the fuel based on TID-14844[1/] release fractions. Using these fission product release fractions available for leakage from the containment, and assuming ground release, conservative meteorological diffusion parameters and design data on the containment sprays, we calculated potential doses at the exclusion boundary and the low population zone radius. Utilizing conservative values for drop size spectrum and deposition velocity and the specific characteristics (e.g., droplet size, flow rate, fall distance, terminal velocity of drop) of the Russellville plant's iodine removal system, we have calculated that iodine removal factors of 4.1 for the 2-hour dose and 10 for the 30-day dose are achievable by the sprays. These dose reduction factors assume as much as 10% of the iodine in

1/ TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, DiNunno, J. J., et al, March 23, 1962.

the containment is in "nonremovable" (organic) form. Allowing these dose reduction factors for iodine removal, the potential 2-hour doses at the exclusion area boundary (0.65 miles) are 4 rem whole body and 210 rem to the thyroid and the 30-day doses at the low population zone radius (4 miles) are about 2 rem body and 81 rem to the thyroid. The applicant has stated that analytical and experimental work on the efficiency of chemical additive sprays is being conducted by B&W, Oak Ridge National Laboratory and others. In addition to sodium thiosulfate, other chemical solutions are also being evaluated. In the event that the results of these development programs indicate that the spray systems might not be as effective as anticipated, the applicant has stated that space will be reserved in the plant so that charcoal adsorber units can be added to further reduce the iodine concentration in the containment.

3.8 Design Conformance to AEC General Design Criteria

The applicant has assessed the Russellville Nuclear Unit design with respect to conformance with the Commission's General Design Criteria published in the Federal Register on July 11, 1967. We have evaluated the application for conformance with the revised criteria and have concluded that the preliminary design of the proposed unit conforms to the intent of these criteria. Recognizing that the proposed criteria, as revised, may be further modified as a result of comment by interested parties, and that the final design may differ somewhat from the preliminary design, we intend to review the proposed unit for conformance to the General Design Criteria again at the operating license stage.

3.9 Emergency Plans

The scope of emergency planning by the applicant, including proposed preparation of written procedures covering reasonably foreseeable emergency operating conditions, is acceptable. Detailed emergency plans for the low population zone will be developed by the applicant in cooperation with state and local authorities. We will evaluate these plans at the operating license review stage.

4.0 RESEARCH AND DEVELOPMENT

There are a number of areas related to pressurized water reactors for which additional research and development will be required. These areas are summarized in this section. We will follow the programs listed below by meeting with the applicant and his contractors and by evaluating reports submitted on these programs. (Expected completion dates are parenthetically noted).

(1) B&W Development of the Emergency Core Cooling System Design

The core cooling research and development being conducted by B&W, must specifically include (a) the completion of the analysis of the spectrum of small break sizes in the loss-of-coolant accident, (b) the development of the analytical techniques for determining blowdown forces on reactor internals, and (c) demonstration that the injection coolant will cool the core including cc sideration of core bypass or formation of a vapor lock. Experimental vibration tests will also be performed to show that induced-vibration will not unseat the core barrel vent valves. (July 1969).

(2) B&W Development of Final Reactor Thermal-Hydraulic, Nuclear and Mechanical Design Parameters

Development work to be performed includes the following:

a. Thermal and Hydraulic Programs

The applicant has proposed scaled flow distribution tests on the vessel and internals and rod bundle tests to determine local mixing and flow effects. This further experimental and analytical work must be done to determine the limiting heat fluxes at various positions within the fuel bundle if the design is to be based on the B&W heat transfer data. (prior to 1969)

b. Fuel rod failure mechanisms during loss-of-coolant-accident (LOCA). Various failure modes of the fuel rods during the LOCA, such as clad melting, eutectic formation, bulging, splitting, or brittle failure, will be examined in an experimental program to assure the continue core cooling capability during a LOCA. (late 1969).

c. High burnup fuel tests

Fuel specimens will be tested at heat rates ranging up to 21.5 kw/foot, burnup ranging up to 75,000 MWD/MTU, and with cladding surface temperature of 650°F. (June 1970).

d. Xenon oscillations

The applicant will further develop analytical techniques to determine the stability margins with respect to xenon oscillations (late 1969). If the stability margins are found to be insufficient, a system for stabilizing and controlling the oscillations will also have to be developed. Results from physics tests on Duke Power Company's Oconee Unit 1 will be used to confirm the analytical results. (2nd quarter 1971).

(3) B&W Control Rod Drive Unit Tests

The prototype tests are being conducted on the B&W control rod drive units under operating temperature, pressure, flow and water chemistry and should provide design adequacy information on the operability and reliability of the system. (prior to 1969).

(4) B&W In-Core Neutron Detectors Tests

The self-powered in-core neutron detectors, which have been developed by B&W, are currently under life testing at B&W's Lynchburgh facility and at the Big Rock Point Nuclear Power Plant. The status of the tests to date are acceptable

(5) B&W Once-through Steam Generator Development and Tests

Investigations of steady-state conditions and operational transients have been completed. Vibrational tests, including vessel response to primary system blowdown, have also been investigated and the thermal response to both primary and secondary blowdown determined. The remaining work involves the development and verification of analytical models for steam system blowdown analyses. (1st quarter 1969).

(6) B&W Development of the Design Details of Iodine Removal System (Chemical Additive to Containment Sprays)

The Russellville plant iodine removal system is being developed by B&W. Chemical characteristics, iodine removal characteristics, compatibility, and radiolysis of spray materials are being evaluated. Experimental investigation of the relationship of absorption rate of containment atmospheric conditions, the effects of process variables on spray nozzle performance and the extent of radiolysis are being conducted by B&W, Oak Ridge National Laboratory, and Battelle Memorial Institute. (early 1969)

4.1 Other Matters to be Further Evaluated During Construction

(1) Instrumentation

There are two areas of instrumentation which will require further information and review.

a. Design of the prompt fuel failure detectors

The applicant has not yet completed the design of these detectors. Upon completion of these detectors, which are to be of two types, one to sample reactor coolant (in the letdown line) and the other to sample containment air, we will review their design capability for adequacy and speed of response as a function of percent of fuel failed.

b. Interaction of control and protection systems

As discussed in Section 3.5 we and the applicant will continue evaluation of the protection and control instrumentation systems with regard to interaction. In particular, we are reviewing the proposed design as it is finalized, for common failure modes, taking into account the possibility of systematic, nonrandom, concurrent failures of redundant devices, not considered in the single-failure criterion.

(2) Containment Design Details

Three containment items have been selected for further evaluation prior to construction of the affected subsystems. This information, which will be developed in the normal course of design, includes the design details and associated analyses for the tendon anchor system and for the liner anchorages.

For tendon anchorages, the applicant has agreed to submit a report giving both predictions and results of the tendon anchorage qualification test. This report, will identify analytical methods and material properties used in the predictions, results of actual tests and comparision of predictions with test results. We plan to review this data as it becomes available as well as additional design information prior to construction of the tendon anchorages.

For liner anchorages, the applicant has agreed to perform tests demonstrating his design will not result in sequential anchorage failures. We plan to review these tests as well as additional design information prior to construction of the liner anchorages.

For Cadweld splices, we and the applicant will agree on the relatively minor changes, if any, required in the statistical sampling strength testing program prior to use of such plices in the plant structures.

(3) Quality Assurance Information

After the constructor has been selected and prior to starting any major construction at the site, we will review the additional quality assurance information, indicated in Section 6.2, which the applicant has agreed to submit.

(4) Reactor Vessel Thermal Shock

As discussed in Section 3.2 we are continuing our review of the problem of thermal shock as a potential consequence of actuation of the core cooling systems.

4.2 Conclusion

We have examined each of the above areas and conclude that they can reasonably be left for later consideration. Moreover, on the basis of the descriptions supplied by the applicant, we conclude that the proposed research and development programs are reasonably designed to resolve the identified safety questions.

5.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards, by letter to Chairman Seaborg, dated September 12, 1968, reported on the Russellville Nuclear Unit. A copy of this letter is attached as Appendix A. The letter contains comments and recommendations which we are implementing, as noted in appropriate sections of this safety evaluation.

The Committee has reiterated its belief that additional consideration be given to common mode instrumentation failures not considered in the single-failure criterion. This is discussed in Section 3.5. The Committee also emphasizes the importance of quality assurance and quality control programs, discussed in Section 6.2; and early training of a sufficient number of personnel for the operating staff, discussed in Section 6.1. Modification of the containment prestressing system design is also mentioned. This is discussed in Section 3.4.

The Committee further calls attention to other matters that warrant careful consideration by the manufacturers of all large, water-cooled, power reactors. These matters, applicable to the Russellville plant involve the following: effects of blowdown forces on primary system components, effects of fuel clad perforation on emergency core cooling performance, and fuel element performance under operational transients, all of which are addressed in Sections 3.2, 3.3 and 4.0 of this report. Additional matters about which the Committee expressed concern include pressure vessel shock from cold water injection, discussed in Section 3.2; prompt detection of gross failure of a fuel element, discussed in Section 3.5; and primary system quality assurance, discussed in Section 6.2. These items will be resolved to our satisfaction as the design work progresses and will be reviewed by the ACRS prior to issuance of an operating license.

The report of the ACRS concluded, ..."The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

6.0 TECHNICAL QUALIFICATIONS OF THE APPLICANT

6.1 Technical Qualifications

We have reviewed the application with respect to the technical qualifications of the Arkansas Power and Light Company (AP&L) and its contractors to design and construct the proposed facility. AP&L has over 45 years experience covering design, construction, and operation of conventional steam, hydro, and diesel electric generating plants which, at the end of 1967, had a total capacity of 1,734 megawatts.

Officers and engineering personnel of AP&L have had previous nuclear experience through AP&L's participation, as a member of the Southwest Atomic Energy Associates, in the Southwest Experimental Fast Oxide Reactor Facility, SEFOR, and through AP&L's participation in the Peach Bottom Atomic Power Station project.

AP&L will rely upon its architect-engineer, contractors, and consultants for technical support during the design and construction of the plant. The Bechtel Corporation has been retained as the architect-engineer and will be responsible for procurement and management of construction of the plant.

Bechtel has wide experience as architect-engineer and engineer-constructor for several pressurized water reactor power plants as well as other types of nuclear and conventional power plants. Babcock and Wilcox will supply the nuclear steam supply system and two fuel cores. B&W has extensive background in supplying nuclear steam supply systems. The turbine generator and its auxiliaries will be supplied by the Westinghouse Electric Corporation.

The number of people proposed for operation of the plant totals 61. Personnel assigned to the plant will have extensive experience in conventional power plants and all supervisory and operating personnel will be given special nuclear training including operator training at a comparable nuclear power plant. The applicant has planned for four-man operating shifts consisting of a shift supervisor with a Senior Operator's License, a plant operator and an assistant plant operator, each with an Operator's License and an auxiliary operator who may have an Operator's License.

On the basis of our review, we conclude that the applicant and its principal contractors have the technical competence to design and build the Russellville Nuclear Unit. We believe, however, that 4-man operating shifts may prove inadequate. We will pursue this matter further with the applicant as it develops its emergency and normal operating procedures and will satisfy ourselves that its training program will assure timely availability of adequate operating manpower.

6.2 Quality Assurance and Quality Control

We have reviewed the quality assurance and control program proposed for the Russellville facility. At our request, the applicant has supplemented its PSAR with additional information which is provided in Supplement 3 (answers to Questions 8.1 through 8.11, 9.5 and 9.7) and in Supplement No. 9.

The applicant's Safety Review Committee reviews all plant designs, specifications, and procedures to ensure compliance with all plant design criteria, codes and standards as set forth in the PSAR with responsibility and authority to reject those which are not in compliance. The AP&L Manager of Safety, who reports directly to the Executive Vice President, is a member of this committee.

AP&L also has established a Quality Assurance Committee (QAC) for the Russellville plant. A key member of the QAC is the Chief Quality Control Coordinator who will be in residence full time at the plant site during construction. He will work closely with the Bechtel Quality Assurance Engineer, who will also be onsite during construction. The Chief QC Coordinator will review all inspection and test procedures prior to inspection or test, monitor tests and inspections at the site and at vendor facilities on a frequent "spot-check" basis and review the results of all quality control programs. The QC Coordinator will be assisted in his duties by AP&L Engineering or Production Department personnel experienced in plant design and construction. In areas where AP&L does not now have experienced personnel, they will either hire or obtain the services of such personnel through a consultant firm.

In addition to the applicant's organization, Bechtel will have a Quality Assurance Engineer (QAE) under the Project Engineer and a separate field inspection force under a Job Engineer. The QAE will have access to and will review, for compliance with established requirements, all Bechtel and vendor quality control procedures and reports of all tests and inspections performed by others in vendors' plants and at the job site. The Bechtel field inspection force reports through the Job Engineer and Project Superintendent to the San Francisco Office Construction Manager while the QAE reports through the Project Manager and Nuclear Power Engineering Manager to the San Francisco Engineering Manager. Bechtel will also have independent checks on quality assurance during the design and pre-fabrication phase by having design bases, designs, and procurement documents, which are prepared by the Project Engineer's staff, reviewed by the staffs of Chief Engineers in each engineering specialty. These Chief Engineers independently report directly to the San Francisco Office Manager Engineering.

B&W, the nuclear steam supply system vendor, has recently established in July of 1968 a quality assurance organization which will be responsible for quality assurance of B&W's nuclear product line from bid proposal to final customer acceptance. This organization, which is independent from the previously existing B&W design, production, and quality control groups, reports directly to the Vice President of the B&W Nuclear Power Generation Department and is responsible for assuring that the Russellville nuclear steam supply system furnished by B&W conforms to all established requirements.

Upon selection of the general contractor for construction of the Russellville facility, the applicant has agreed to submit the following information: (a) a list of all organizations involved in the design and construction of this plant, (b) description of the various responsibilities of all organization including quality assurance and control, (c) a schedule of major construction activities, (d) a listing of responsible persons (plant site and vendor shops) as contacts for Division of Compliance inspectors, (e) location of complete specifications and quality assurance and control documents, and (f) a list of all major vendor shop locations.

Subject to our review of this additional information, we conclude that the applicant, together with its contractors, will have an adequate quality assurance program and that independent checks on quality assurance and quality control can be provided at all stages, from establishing adequate design bases initially, through design, fabrication, testing and final inspection.

7.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application to suggest that the applicant is owned, controlled or dominated by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the regulations. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

8.0 CONCLUSIONS

On the basis of the proposed design of the Arkansas Power and Light Company's Russellville Nuclear Unit; the criteria, principles, and design arrangements for systems and components thus far described, which include all of the important safety items; the calculated potential consequences of routine and accidental release of radioactive materials to the environs; the scope of the development program which will be conducted; and the technical competence of the applicant and the principal contractors; we have concluded that the appropriate findings as set forth in the notice of hearing of this proceeding, September 20, 1968, can be made by the Director of Regulation.

In summary, we conclude that the proposed plant can be built and operated at the proposed location without undue risk to the health and safety of the public.

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SEP 1 2 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON RUSSELLVILLE NUCLEAR UNIT

Dear Dr. Seaborg:

At its one-hundred-first meeting, September 5-7, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Arkansas Power and Light Company to construct the Russellville Nuclear Unit. This project had been considered previously during Subcommittee meetings on August 23, 19[illegible], at the site, and on September 4, 1968, in Washington, D. C. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Arkansas Power and Light Company, the Bechtel Corporation, the Babcock and Wilcox Company, and the AEC Regulatory Staff. The Committee also had available the documents listed.

The plant will be located about six miles from Russellville, Arkansas, on a peninsula formed by the Dardanelle reservoir. The normal elevation of the reservoir is controlled downstream by the Dardanelle Lock and Dam No. 10 on the Arkansas River. An emergency reservoir on the site will provide adequate storage of water in the unlikely event of failure of Lock and Dam No. 10. The consequences of the maximum probable flood have been studied, and adequate protection has been provided for the critical equipment of the nuclear unit.

The proposed nuclear unit is a pressurized water reactor, 2452 MWt and 850 MWe, and is similar to previously approved units (e.g., Rancho Seco, Crystal River, and Three Mile Island, ACRS Reports of July 19, 1968, May 15, 1968, and January 17, 1968, respectively). The Committee continues to call attention to matters that warrant careful consideration by the manufacturers of all large, water-cooled, power reactors.

The Committee reiterates its belief that the instrumentation design should be reviewed for common failure modes, taking into account the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. The applicant should show that the proposed interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner, considering the possibility of systematic component failure. The Committee believes this matter can be resolved with the Regulatory Staff.

The containment for the reactor is a prestressed concrete vessel similar to previously approved designs (e.g., Rancho Seco), but with modification of the prestressing system design.

The Committee emphasizes the importance of the implementation and management of the quality assurance and quality control programs necessary to achieve the design, construction, and operation objectives.

Inasmuch as a long lead time is required in the training of the operating staff, the Committee emphasizes the need for early training of sufficient personnel to assure adequate operating manpower.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Original signed by
Carroll W. Zabel
Carroll W. Zabel
Chairman

References Attached.

References - Russellville Nuclear Unit

1. Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
2. Volume I - Preliminary Safety Analysis Report, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
3. Volume II - Preliminary Safety Analysis Report, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
4. Supplement No. 1 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated January 22, 1968.
5. Supplement No. 2 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated February 14, 1968.
6. Supplement No. 3 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated May 3, 1968.
7. Supplement No. 4 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated June 5, 1968.
8. Supplement No. 5 to the Arkansas Power and Light Company Preliminary Safety Analysis Report, dated July 3, 1968.
9. Corrections to Supplement No. 5 to the Arkansas Power and Light Company Preliminary Safety Analysis Report, dated July 10, 1968.
10. Supplement No. 6 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated July 11, 1968.
11. Correction to Supplement No. 6 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated July 15, 1968.
12. Supplement No. 7 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 15, 1968.
13. Supplement No. 8 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 26, 1968.
14. Supplement No. 9 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 30, 1968.

APPENDIX B

CHRONOLOGY

REGULATORY REVIEW OF THE ARKANSAS POWER AND LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT

1.	November 29, 1967	Submittal of Preliminary Safety Analysis Report and License Application.
2.	January 22, 1968	Submittal of Supplemental No. 1, response to AEC General Design Criteria.
3.	January 24, 1968	Meeting with applicant to discuss plans and scheduling of regulatory review.
4.	February 14, 1968	Submittal of Supplement No. 2, design changes in electrical systems and emergency core cooling systems, and data on Dardenell Lock and Dam.
5.	February 28, 1968	Meeting with applicant to discuss areas of the Preliminary Safety Analysis Report that require additional information.
6.	April 3, 1968	Request to applicant for additional information on site, safety analysis, reactor, instrumentation and control, emergency power, engineered safety features quality assurance, training schedules, emergency plants, and initial tests and operations.

7.	May 6, 1968	Request to applicant for additional information on Foundation and Structural Design and miscellaneous other items.
8.	May 3, 1968	Submittal of Supplemental No. 3 in response to April 13, 1968 request for additional information.
9.	May 17, 1968	Meeting with applicant to discuss training schedules and operating staff.
10.	June 5, 1968	Submittal of Supplement No. 4 in response to May 6, 1968 request for additional information.
11.	June 20, 1968	Meeting with the applicant to discuss modified containment design proposed by applicant, site matters and other areas.
12.	July 3, 1968	Submittal of Supplement No. 5, changes in containment design.
13.	July 11, 1968	Submittal of Supplement No. 6, supplemental information in clarification of areas discussed at June 20, 1968 meeting.
14.	August 6, 1968	Meeting with applicant to discuss containment design matters

15.	August 15, 1968	Submittal of Supplement No. 7, supplemental information in clarification of areas discussed at August 16, 1968 meeting.
16.	August 23, 1968	ACRS Subcommittee meeting and Russellville site visit.
17.	August 26, 1968	Submittal of Supplement No. 8, additional supplementary information in clarification of containment design.
18.	August 27, 1968	Meeting with applicant to discuss quality assurance and quality control plans and organizations.
19.	August 30, 1968	Submittal of Supplement No. 9, documenting information and oral commitments given at August 27 meeting and miscellaneous other items.
20.	September 3, 1968	Meeting with applicant to discuss containment liner anchorage design.
21	September 4, 1968	ACRS Subcommittee meeting.
22.	September 5, 1968	ACRS meeting.
23.	September 6, 1968	Submittal of Supplement No. 10, updating financial and personnel information and correcting minor errors.
24.	September 12, 1968	ACRS Report issued.

APPENDIX C1

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

MAY 29 1968

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in reply to Mr. Boyd's letter of December 11, 1967, requesting our comments on the application by the Arkansas Power and Light Company for construction permit for the proposed Russellville Nuclear Unit, Pope County, Arkansas, AEC Docket No. 50-313.

The project would be located on a 1,100-acre site on a peninsula at Dardanelle Reservoir, Pope County, Arkansas. A pressurized water reactor would be used as a power source and the plant is designed for an ultimate output of 2,568 thermal (880 gross electrical) Mwt. Cooling and dilution water will be withdrawn from a small inlet embayment west of the plant at a rate of approximately 1,700 c.f.s. and be discharged into the large Illinois Bayou embayment east of the plant, after receiving radioactive and heat wastes. As currently designed, the temperature of the cooling water would be raised approximately 15° at the condenser when the plant is operating at full capacity. The applicant is cooperating with the Fish and Wildlife Service and the Arkansas Game and Fish Commission in the development of an environmental surveillance program.

Dardanelle Reservoir, especially the Illinois Bayou embayment, supports valuable fish and wildlife resources. The large embayment is a productive nursery and harvest area for fish. Waterfowl make extensive use of the reservoir for resting during the migration period. Public and private use facilities on Federal and private land around the embayment are highly developed. Indications are that future development around the embayment will probably result in higher recreational use there than any comparable area of the reservoir. Sport fishing is presently, and will continue to be, one of the chief recreational use attractions in the embayment. Commercial fishing is limited but moderately valuable.

The application indicates that the release of radioactive wastes would not exceed maximum permissible limits prescribed under the Code of Federal Regulations. Although these limits refer to maximum levels of radioactivity that can occur in drinking water for man without resulting in any known harmful effects, operations within these limits may not always guarantee that fish and wildlife will be protected from adverse effects.

If concentrations in receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required, but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides.

In view of the above, we believe that the environmental monitoring program planned by the applicant should include pre- and post-operational radiological monitoring of selected organisms which require the waste elements or similar elements for their metabolic activities. These surveys should be planned in cooperation with the Fish and Wildlife Service and the appropriate Federal and State agencies.

In view of the extensive sport fishery and the potential value of the commercial fishery in the project area, it is imperative that every possible effort is to be made to protect the valuable resources from radioactive contamination. Therefore, it is recommended that the Arkansas Power and Light Company be required to:

1. Include in their pre-operational environmental surveillance program radiological monitoring of water and sediment samples and of organisms indigenous to the project area that concentrate and store radioactive isotopes. Water and sediment samples should be collected within 500 feet of the reactor effluent outfall site and be measured for gamma radioactivity. Aquatic plants, mollusks, crustaceans and fish should be collected as near as possible to the reactor effluent outfall site and be analyzed for both beta and gamma radioactivity.

2. Prepare a report of pre-operational radiological monitoring and provide five copies to the Secretary of the Interior for evaluation prior to project operation.

3. Continue a radiological monitoring program similar to that specified in recommendation 1 above, analyze the data, and prepare and submit reports every six months during reactor operation or until it is conclusively demonstrated that no significant adverse conditions exist. Five copies of these reports should be submitted to the Fish and Wildlife Service for distribution to the appropriate State and Federal agencies for evaluation.

4. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to acceptable levels if it is determined by the monitoring program that the release of radioactive effluent might result in harmful concentrations of radioactivity in fish and wildlife.

We understand that the Commission's regulatory authority over nuclear power plants involves only those hazards associated with radioactive materials. However, we recommend and urge that before a construction permit is issued, the possibility of thermal and other detrimental effects on fish and wildlife which may result from plant construction and operation be called to the applicant's attention.

We are concerned particularly with the possibility of damages to aquatic life from the heated effluent. Large volumes of heated water discharged into an aquatic environment may not only be detrimental to fish directly, but may also affect these resources indirectly through changes in the environment. The proposed heat load may adversely affect fish habitat and productivity in the Illinois Bayou embayment during the periods (spring and summer) when fish reproduce and have a maximum growth rate. It is likely that the use of the area for spawning will be greatly reduced. It is likely that fish will disperse and avoid the heat-affected area during the maximum temperature months of June through September. Conversely, it is expected that fish will be attracted to the discharge channel and heat-affected area during winter months, resulting in high fisherman-use there.

A General Plan for use of project lands and waters for wildlife conservation and management has been approved for Dardanelle Reservoir by the Secretary of the Army, the Secretary of the Interior, and the Director of the Arkansas Game and Fish Commission. The Russellville Nuclear Unit would occupy land and water covered, in part, by the General Plan. The General Plan provides for a subsequent management agreement between the Department of the Army and the Arkansas Game and Fish Commission. It further provides that the subsequent agreement may make adjustments in the boundaries of the areas shown in the General Plan by the addition or deletion of tracts mutually agreed upon by the parties making the agreement. We understand that the Department of the Army and the Arkansas Game and Fish Commission are now negotiating an agreement pursuant to the General Plan. The Company should be made aware of these documents and plan its operations so that they are in accordance with the Arkansas Game and Fish Commission's fish and wildlife management plan for the reservoir.

The applicant has given assurance that additional studies will be carried out, and has to date cooperated fully with the Fish and Wildlife Service and the Arkansas Game and Fish Commission in discussing and developing plans for the protection of fish and wildlife in the area. This study program should complement the radiological monitoring program recommended above, should be designed to measure habitat changes in the affected area of Dardanelle Reservoir, and should be carried out prior to and during plant operation, so that comparative data will be available for analysis.

In view of the above, we recommend that the Atomic Energy Commission urge the Arkansas Power and Light Company to:

1. Continue to cooperate with the Fish and Wildlife Service, Arkansas Game and Fish Commission, and other interested Federal and State agencies in developing plans for ecological surveys, initiate these studies at least two years before reactor operation, and continue them during project operation on a regular basis or until it has been conclusively demonstrated that no significant adverse conditions exist.

2. Meet with the above-mentioned Federal and State agencies at frequent intervals to discuss new plans and to evaluate results of the ecological surveys.

3. Make such modifications in plant structures and operations, including but not limited to facilities for cooling discharge waters, as may be determined necessary to protect the fish and wildlife resources of the area.

The opportunity to present our views is appreciated.

Sincerely yours,

Clarence F. Pautzke

Commissioner

APPENDIX C2

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

IN REPLY REFER TO:

AUG 29 1968

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in response to Mr. Boyd's letter of July 16 transmitting Amendment No. 6, dated July 11, 1968, to the application by Arkansas Power and Light Company for a construction permit for the proposed Russellville Nuclear Unit, Pope County, Arkansas, Docket No. 50-313.

Modification of project plans to reverse the direction of cooling water flow through the project would not alter overall effects of the project on fish and wildlife significantly. The recommendations contained in our letter of May 29 are still applicable.

Thank you for the opportunity for comments on Amendment No. 6.

Sincerely yours,

Clarence F. Pautzke
Commissioner

UNITED STATES GOVERNMENT

Memorandum

TO : Peter A. Morris, Director
Division of Reactor Licensing

DATE: JAN 22 1968

FROM : Milton Shaw, Director
Division of Reactor Development & Technology

SUBJECT: SAFETY ANALYSIS REPORTS

RDT:NS:S349

Reference is made to the letters of November 22, 1967, December 11, 1967, and December 26, 1967, from the Division of Reactor Licensing, to the Environmental Science Services Administration requesting comments on the following safety analysis reports respectively:

Rancho Seco Nuclear Generating Station Unit No. 1
Sacramento Municipal Utility District
Preliminary Safety Analysis Report
Volumes I, II, III and IV dated November 1967

Russellville Nuclear Unit
Arkansas Power and Light
Preliminary Safety Analysis Report
Volumes I and II dated November 29, 1967

Donald C. Cook Nuclear Plant
Indiana and Michigan Electric Company
Preliminary Safety Analysis Report
Volumes I, II and III dated December 18, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and their comments are attached.

Attachments:
Three Sets of Comments (Orig. & 1 copy)

APPENDIX D

Comments on

Russellville Nuclear Unit
Arkansas Power and Light
Preliminary Safety Analysis Report
Volumes I and II dated November 29, 1967

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
January 10, 1968

The analysis of the Fort Smith and Little Rock meteorological data indicates that a continental diffusion climate can be expected at the Russellville site. This means a pronounced difference between daytime and nighttime atmospheric diffusion rates, with the lower wind speeds and slower diffusion occurring at night. The predominant daytime wind direction for the general area would be from the southwest as shown by the Little Rock wind rose. Nighttime wind directions with inversion conditions will most likely be towards the Dardanelle Reservoir of the Arkansas River.

The analysis of the Little Rock hourly weather reports with regard to diffusion types shows an average frequency of about 35% for Pasquill F condition during the four months considered (see Table 2A.15). The annual nighttime wind speeds were less than 3 knots about 20% of the time at Little Rock (see Table 2A.6). On this basis, it would seem appropriately conservative to use inversion diffusion conditions (Type F) and a 1 m/sec wind speed to compute the initial two-hour average concentration. This would result in a concentration of 6.4×10^{-4} sec m^{-3} at the site boundary assuming a ground source with no credit for building-induced dilution. Taking credit for the building effect as determined empirically in tests at the National Reactor Testing Station would result in a concentration value of about 2×10^{-4}, which agrees with the applicant's value.

The analysis of the persistence of a diffusion condition in a unidirectional flow (Tables 2A.17 and 18) shows that no cases persisted longer than 10 hours. Consequently, for the 24-hour average concentration it would be conservative to assume inversion conditions, a 2 m/sec wind with concentrations averaged over a 22 1/2 degree arc. At the site boundary this would result in an average concentration of 7×10^{-5} sec m^{-3}, which is in reasonable agreement with the applicant's computation.

In summary, a reasonable, conservative analysis has been made of the atmospheric diffusion conditions of the Russellville site which provides a sound basis for a preliminary safety evaluation of the proposed nuclear plant.

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

DOCKET NO. 50-313

AUG 16 1968

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger S. Boyd is a review of geologic and hydrologic aspects of the site for the Russelville Nuclear Station proposed by the Arkansas Power and Light Company.

The review was prepared by H. H. Waldron and E.L. Meyer and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

Arthur A Baker

Acting Director

Enclosure

Russelville Nuclear Unit
Pope County, Arkansas
AEC Docket 50-313

Hydrology

The site is located on the left bank of the Arkansas River 6 miles upstream from Dardanelle Lock and Dam No. 10. The plant site grade at 353 feet msl (above mean sea level) is 15 feet above the normal operating level of Dardanelle Reservoir.

Flood stages in the pool of Dardanelle Reservoir for a computed maximum probable flood of 1,500,000 cfs (cubic feet per second) have been given by the Corps of Engineers as 353 feet msl at Dardanelle Dam and 389.5 feet msl at the upstream end of the reservoir. The applicant's estimate of 358 feet msl for the stage of such a flood at the site appears reasonable. The failure of Ozark Dam about 46 miles upstream from the site during such a flood could cause an additional rise in stage. The head differential across Ozark Dam during a maximum probable flood as computed by the Corps of Engineers would be 11.5 feet, and on that basis the applicant has estimated an additional 3 feet rise at the site resulting in a stage of 361 feet. This appears to be reasonable.

At a stage of 361 feet the site grade would be overtopped by 8 feet and the reactor structures would be surrounded by water. A certain amount of wave action may then be expected and should be reflected in the level of flood protection chosen for essential equipment.

The cooling water requirements of the reactor are given as 1,700 cfs (cubic feet per second). Flow of the Arkansas River has been measured at a gage at Dardanelle 6 miles downstream from the site. Average flow during 1937-66 was 34,920 cfs; minimum flow was 416 cfs, and the lowest mean monthly flow was 592 cfs in October 1956. Low flow occurs generally in late summer and fall.

Geology

The analysis of the geology of the Russellville Nuclear Generating Plant in Arkansas, as presented in AEC Docket No. 50-313 and supplements, was reviewed and compared with the available literature. The analysis appears to be carefully derived and to present an adequate appraisal of those aspects of the geology that would be pertinent to an engineering evaluation of the safety of the site.

There are no identifiable active faults or other recent geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site.

Tectonically the site is located near the axis of the Scranton syncline, one of several westward-trending, gentle folds that characterize the

Arkoma Basin--a major structural and topographic feature of Arkansas and eastern Oklahoma that developed in late Paleozoic time. Although several ancient faults are associated with the Arkoma Basin folded structures in the area, none of these appears to have been tectonically active since latest Paleozoic time.

The limited subsurface data available indicate that the major units of the nuclear facility will be founded on a hard, dense shale (the McAlester Formation), which should provide an adequate foundation for the proposed structures.

(60)

APPENDIX F

U.S. DEPARTMENT OF COMMERCE
ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
ROCKVILLE, MD. 20852

AUG 15 1968

IN REPLY REFER TO: C23

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of Russellville, Arkansas, and vicinity. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity of the area as presented by the Arkansas Power and Light Company in the "Preliminary Safety Analysis Report," and we are now submitting our conclusions on the seismicity factors.

If we may be of further assistance to you, please do not hesitate to contact us.

Sincerely yours,

James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

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U.S. ATOMIC ENERGY COMM.
MAIL & RECORDS SECTION

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2860

REPORT ON THE SITE SEISMICITY FOR THE RUSSELLVILLE NUCLEAR UNIT, ARKANSAS

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has examined the seismicity of the area around the proposed site near Russellville, Arkansas, and has examined a similar analysis made by the applicant, the Arkansas Power and Light Company in the "Preliminary Safety Analysis Report." The applicant's report is satisfactory for an evaluation of the seismic factor of the site.

Based upon the review of the seismic history of the site and the surrounding area and the related geologic conditions, the Coast and Geodetic Survey agrees with the applicant that an acceleration of 0.10 g on good foundation would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. In addition, the Survey agrees with the applicant that the acceleration of 0.20 g would represent the ground motion from the maximum earthquake likely to affect this site. We believe this value would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

August 14, 1968

APPENDIX G

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

19 August 1968

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
The Russellville Nuclear Unit, Arkansas Power and Light Company
(AEC Docket No. 50-313)

Dear Dr. Morris:

We are transmitting herewith two copies of our report entitled "Adequacy of the Structural Criteria for the Russellville Nuclear Unit," prepared by Drs. W. J. Hall, W. H. Walker and myself.

Sincerely yours,

N M Newmark

N. M. Newmark

mlw
cc: W. J. Hall
W. H. Walker
Enclosure

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO AEC REGULATORY STAFF

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE RUSSELLVILLE NUCLEAR UNIT

ARKANSAS POWER AND LIGHT COMPANY

(AEC Docket No. 50-313)

by

N. M. Newmark, W. J. Hall and W. H. Walker

August 1968

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE RUSSELLVILLE NUCLEAR UNIT

Arkansas Power and Light Company

by

N. M. Newmark, W. J. Hall and W. H. Walker

INTRODUCTION

This report is concerned with the adequacy of the containment structures and components for the Russellville Nuclear Unit for which application for a construction permit has been made to the U. S. Atomic Energy Commission by the Arkansas Power and Light Company. The facility is located on a peninsula in the Dardanelle Reservoir, Arkansas River, Pope County, Arkansas, about 6 miles WNW of Russellville, and 2 miles SE of London, Arkansas.

Specifically this report is concerned with the design criteria that determine the ability of the containment system and Class I equipment and piping as well as Class II structures and equipment, to withstand an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration simultaneously with the other loads forming the basis of the design. The facility also is to be designed to withstand a Design Basis Earthquake of 0.20g maximum horizontal ground acceleration to the extent of ensuring safe shutdown and containment.

This report is based on information and criteria set forth in the Preliminary Safety Analysis Reports (PSAR) and supplements thereto listed at the end of this report. Also, we have participated in discussions with the applicant and the AEC Regulatory Staff concerning the design of this unit.

DESCRIPTION OF FACILITY

The Russellville Nuclear Unit is described in the PSAR as consisting of a pressurized-water type reactor employing two closed cooling loops connected in parallel to the reactor vessel. The system is arranged as two heat transport loops, each with two circulating pumps and one steam generator; one of the loops contains an electrically heated pressurizer. The nuclear steam supply system will be furnished by the Babcock and Wilcox Company, and the turbine generator is to be supplied by the Westinghouse Electric Corporation. The plant is to be designed for a power level of 2452 MWt (850 MWe).

The reactor containment structure is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system of horizontal and vertical tendons. The dome is post-tensioned using a 3-way system. The hoop tendons are to be placed in three 240° systems using three buttresses as anchorages, with the tendons staggered so that half of the tendons at each buttress terminate at that buttress. The foundation slab is conventionally reinforced with high-strength reinforcing steel.

The cylinder has an internal diameter of 116 ft. and an inside height of 206 ft. The distance from the top of the foundation slab to the springline of the domed roof is approximately 166 ft. The vertical wall thickness is noted to be ft. - 9 in. and the dome thickness, 3 ft. - 3 in. The foundation slab thickness is about 9 ft.

For prestressing, the applicant proposes to use 90 to 184 wire tendons, unbonded. The discussion presented in the PSAR suggests that the BBRV type anchorage system will be employed, although the PSAR notes that other prestressing systems will continue to be studied. The prestressing tendons will be protected against corrosion by a pressure-injected casing filler. The liner plate will conform to specification ASTM-A442, Grade 60, and will be 1/4 in. in thickness. The reinforcing steel in the base slab of the containment structure will conform to ASTM designation A432-65; this steel possesses a minimum yield strength of 60,000 psi. Splices in bars larger than No. 11 will be made by the Cadweld method.

The design of the containment structure for this facility is essentially similar to that employed for the Rancho Seco Nuclear Generating Station Unit No. 1.

The geological description of the site indicates a stiff clay and silty clay of 13 to 23 foot thickness overlying hard and dense horizontally bedded shale of the Pennsylvanian McAlester formation. All major structures of the facility will be founded on the underlying McAlester formation shale bedrock. No active or recent faulting has been mapped in the area of the proposed site. The closest known faults are the London and Perry View faults located 5 or 6 miles from the site.

SOURCES OF STRESSES IN CONTAINMENT STRUCTURES IN CLASS I COMPONENTS

The reactor containment structure is to be designed for the following loadings and conditions: dead load; live load (including snow and equipment loads); prestressed loadings; design accident temperature of about 285°F and pressure of 59 psig; an air test pressure of 115 percent of the design pressure; an external pressure loading with a differential of approximately $2\frac{1}{2}$ psi from outside to inside; wind loading corresponding to 80 mph basic wind at 30 ft. above grade; buoyancy loadings; tornado loading associated with a 300 mph tangential wind velocity and a 40 mph forward progression velocity, including a differential pressure of 3 psi from inside to outside with associated missiles, and earthquake loading as described next.

The seismic design is to be made for an Operating Basis Earthquake based upon a maximum horizontal ground acceleration of 0.10g and a Design Basis Earthquake based upon a maximum horizontal ground acceleration of 0.20g.

The containment walls and liner are shielded by various types of barriers from impact from missiles which possibly could have enough energy to strike or penetrate them. The high-pressure reactor cooling system equipment which could be the source of missiles is screened either by the containment shield wall enclosing the reactor cooling loops, by the concrete operating floor, or by a special missile shield to block any passage of missile to the containment walls.

The general criteria controlling the design of piping and reactor internals for seismic loadings are presented in various places in the PSAR.

COMMENTS ON ADEQUACY OF DESIGN

Foundations and Dams

The major facility structures are to be founded directly on competent bedrock, and on the basis of the information presented in the PSAR and amendments, the foundation conditions appear acceptable to us.

The Dardanelle Reservoir from which the plant will draw its cooling waters is discussed in several places in the PSAR and particularly in Appendix 2F and in the answers to Questions 2.7 and 2.8 of Supplement No. 3. The analysis of the Dardanelle Lock and Dam as reported in Appendix 2F suggests that some damage to the Lock and Dam facility might be expected. Thus, the applicant notes in the answer to Question 2.7 that emergency shutdown cooling water will be supplied from an emergency reservoir to be located northwest of the plant site. The emergency reservoir will be excavated in impervious clay and will have an effective storage capacity of about 35 acre feet. We concur in this approach for an assured source of cooling water in view of the possible effects of an earthquake on the Dardanelle Lock and Dam.

The effect of a flood on the structure is discussed in the answer to Question 2.8 of Supplement No. 3. It is noted there that the plant grade level is elevation 353 ft. and the maximum elevation of a flood is estimated to be 361 ft. The applicant indicates that the early forecast of a severe flood of this type would provide ample time for precautionary measures in terms of plant shutdown. All Class I equipment is either located above maximum probable flood level or protected by waterproof Class I structures which are designed for buoyancy effects.

Gas Pipeline

In the answer to Question 2.11 of Supplement No. 3, there appears a discussion of the natural gas transmission pipeline which crosses the discharge water channel. It is indicated in the answer to that question that the existing pipeline crossing will be re-layed beneath the water channel with 4 ft. of earth cover. We understand that it will be possible to valve off this section of line in the event of difficulty. It is noted that the pipeline will be at its closest about 400 ft. from the intake structure and 600 ft. from the containment structure. These distances are sufficient, we believe, to preclude any serious consequences with regard to plant safety in the event of a pipe rupture.

Seismic Design and Criteria

We are in agreement with the earthquake loading criteria selected for the seismic design, namely that associated with an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration and a Design Basis Earthquake of 0.20g maximum horizontal ground acceleration. These earthquake design criteria are in agreement with those given by the U. S. Coast and Geodetic Survey (Ref. 2).

The response spectra for the Operating Basis Earthquake and Design Basis Earthquake to be employed in the dynamic analysis are presented as Fig. 5A-1 and 5A-2 of Appendix 5A of the PSAR. These spectra are scaled after those presented in publications by Dr. G. W. Housner, and we concur in their use.

The earthquake analysis will include the effects of vertical earthquake excitation which will be taken as 2/3 of the horizontal component as noted on page 5-3 of the PSAR. It is noted in the answer to Question 12.3.6 that the effects of vertical and horizontal earthquake motions will be combined linearly and directly with each other and with the other applicable stresses. We are in agreement with these design criteria.

The percentage of critical damping to be employed in the analysis is listed on page 5-A-5 of the PSAR, and we are in agreement with the values given there.

The method of dynamic analysis is described in Section 5.1.5.6 of the PSAR. The method of analysis is not described in enough detail to evaluate it completely; however, it would be our recommendation that a standard modal analysis procedure be employed to take account of structural rocking, lateral translation, and the shearing and flexural distortion of the structure. With proper attention to damping and coupling of the various modes, it should be possible to arrive at reasonable and consistent values of direct stress, shear, moment, etc.

The loading combinations to be employed for the design of the containment structure are given in Section 5.1.4 of the PSAR. The loading combination expressions given appear acceptable to us, and it is noted that for these load factor combinations, the resistance will be less than the yield strength of the structure. We concur in this approach.

The design of Class II structures is discussed in the answer to Question 12.3.2 where it is noted that the design of such items will be for Zone 1 of the Uniform Building Code. It would be our recommendation that for critical Class II items that are of special significance in terms of plant safety, the design be made on the basis of about 2/3 of Zone 3 of the Uniform Building Code.

The design approach as outlined for handling principal concrete tension and combined tension and membrane shear appear acceptable to us.

The design of the liner and anchors is discussed in various sections of the PSAR. We are advised that the liner design is still under study and that further information will be forthcoming during the design phases. It is our belief that the liner design can be carried out satisfactorily and adequately, and we can see no particular difficulty here which will preclude going forward with the construction permit.

The general approach outlined for the prestressed design receives attention in various parts of the PSAR and Supplements and other material made available to us. The design for this plant employing three buttresses and 90 to 184 wire tendons is relatively new. The applicant indicates that many factors associated with this post-tensioning are receiving added study, as for example the problems associated with the friction arising from the large pulling arc (240°). On the basis of the information available to us, and realizing that additional studies are underway and will be carried forward during the design phases, we can see no reason why the proposed system will not be acceptable.

The design procedure for handling the statical design, namely use of the finite element technique, coupled with special study and procedures for handling the primary and secondary loading around penetrations, appears satisfactory to us.

Class I Piping, Equipment, Vessels and Reactor Internals

Only general statements are made in the PSAR concerning the design of Class I piping, equipment, vessels and reactor internals. However, the answer to Question 10.4.3 of Supplement No. 3 suggests that the criteria employed for Crystal River Unit 3 of the Florida Power and Light Corporation will be applicable to this plant, and reference is made to the answer to Question 9.11 of the Crystal River application. It is noted that the calculations and design will not be completed until mid 1969.

On the assumption that the approach outlined in the Flori . Power and Light Corporation application for Crystal River Unit 3 will be followed, we concur in the proposed approach.

Controls, Instrumentation, Batteries, etc.

Only general information is noted in the PSAR concerning the seismic design criteria for critical elements of control, instrumentation, batteries, etc. It would be our recommendation that criteria for these items be examined in detail during the design phases, to insure that the items can withstand the forces, motions and tilt that might be associated with an earthquake.

Quality Control and Inspection

The matter of quality control, inspection and acceptance is discussed throughout the PSAR and amendments. The procedures outlined appear acceptable to us.

CONCLUDING COMMENTS

On the basis of the information presented in the PSAR and supplements, and in keeping with the design goal of providing serviceable structures and components with a reserve of strength and ductility, we believe that the design outline for the containment and other Class I structures and equipment and for Class II structures and components can provide an adequate margin of safety for seismic resistance. However, in the body of the report we have offered comments concerning the method of dynamic analysis, and the design criteria for Class II structures. It is understood that studies will continue during the design phases on the design of the liner anchorage and the pre-stressing tendon system, and it is suggested that the seismic design criteria for critical instrumentation be developed and implemented during the design phases.

REFERENCES

1. "Preliminary Safety Analysis Report - Volumes I, II (and Supplements No. 1, 3, 4, 5 and 6)," Russellville Nuclear Unit, Arkansas Power and Light Company, 1968.

2. "Report on the Seismicity of the Russellville Nuclear Unit Site," U.S. Coast and Geodetic Survey, Rockville, Maryland______________.