RETURN TO REGULATORY CENTRAL FILES
ROOM 016

SUPPLEMENT No. 1

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

NUCLEAR POWER PLANT

POPE COUNTY, ARKANSAS

DOCKET NO. 50-313

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Arkansas Power & Light Company (hereinafter referred to as AP&L or the applicant) by application dated November 2 , 1967, and as subsequently amended, requested a license to construct and operate a pressurized water reactor, identified as the Russellville Nuclear Unit (later as Arkansas Nuclear One - Unit 1 and hereinafter referred to as ANO-1) at a site on the Dardanelle Reservoir in Pope County, Arkansas. The Atomic Energy Commission's Regulatory staff reported the results of its review prior to construction in a Safety Evaluation Report dated October 1, 1968. Following a public hearing before an Atomic Safety and Licensing Board in Russellville, Arkansas on October 30, 1968, the Commission issued Provisional Construction Permit CPPR-57 on December 6, 1968.

On April 23, 1971, the applicant filed, as Amendment No. 19, the Final Safety Analysis Report (FSAR) required by 10 CFR 50.34(b) as a prerequisite to obtaining an operating license for the facility. The Atomic Energy Commission reported the results of its operating license review in a Safety Evaluation Report dated June 6, 1973. This supplement to that Safety Evaluation Report documents the conclusions of the Regulatory staff review of fuel densification and matters covered in the report of the Advisory Committee on Reactor Safeguards, and provides additional discussion of the staff's evaluation of high energy line rupture outside containment and electrical review matters.

2.0 FUEL DENSIFICATION

2.1 General

2.1.1 Background

On November 14, 1972, the Regulatory staff issued a report entitled, "Technical Report on Densification of Light Water Reactor Fuels"[(1)*] which resulted from the staff's consideration of the Ginna fuel densification phenomenon. Based upon the findings in this report the staff requested on November 20, 1972 that the applicant provide analyses and relevant bases, in accordance with the densification report,[(1)] that determine the effects of fuel densification on normal operation, transients and accidents for ANO-1. On January 16, 1973 the Duke Power Company filed a response to the request for Oconee Unit 1 as a lead plant for this evaluation.[(2,3)] On March 14, 1973, the staff requested additional information. Duke Power Company filed a response to this request on April 13, 1973.[(4,5)] On June 15, 1973 the applicant filed a response to these requests specifically for ANO -1[(6)].

The staff's technical review of fuel densification as it applies to ANO-1, and the technical evaluation of the applicant's safety analysis of steady state operation, operating transients and postulated accidents taking into account the effects of densification are presented in this section.

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*Numbers in () refer to references listed in Section 6.0.

This evaluation relies upon the July 6, 1973 Regulatory staff report "Technical Report on Densification of Babcock & Wilcox Reactor Fuels"[7] which concluded that B&W's fuel densification models are in compliance with the staff's initial densification report.[1]

The staff has concluded that the operation of ANO-1 for the first cycle at power levels up to 100 percent of full power, in accordance with the Technical Specifications, will not present an undue risk to the health and safety of the public.

2.1.2 Scope of Review

The essential elements that must be considered in evaluating the effects of fuel densification have been set forth in the staff's initial densification report.[1] Since the performance of the facility in steady state operation and during various postulated transients and accidents had been established previously as reported in the Final Safety Analysis Report (FSAR) without the assumption of fuel densification, it was only necessary to evaluate those changes in the analyses and in the results that are attributed to fuel densification. The effects of fuel densification on the steady state operation and on the course of plant transients and postulated accidents were evaluated by the applicant and reviewed by the staff.

The staff reviewed the effects of fuel densification for ANO-1 using the staff's guidelines, the technical evaluation of the applicant's safety analysis of steady state operation, operating transients and postulated accidents and the generic evaluation[7] of B&W methods for assessing fuel dens-fication and its effects. In the evaluation the applicant appropriately considered the staff guidelines including the effects of instantaneous and anisotropic densification (initial density minus 2σ, and final density of 96.5% theoretical density), the assumption of no clad creepdown as a function of core life, and the assumption of an axial gap leading to a power spike. The staff reviewed the effects of fuel manufacturing and reactor operating parameters on the fuel densification mechanism. The generic evaluation of these items is included in Reference 7. The staff reviewed B&W's assumptions, methods, and computer codes used in evaluating the fuel densification effects. The generic evaluation of B&W's models is also included in Reference 7. The mechanical integrity of the fuel cladding and the thermal performance of the fuel were considered in the analyses of steady state operation, operating transients, and postulated accidents as discussed in the following sections.

2.2 Mechanical Integrity of Cladding

Clad creepdown during the core life is not considered by the applicant in the calculation of gap conductance. This is a conservative assumption since the reduced gap size due to clad creepdown would result in a higher gap conductance and thus in a lower stored energy in the fuel. The staff reviewed the B&W method for calculating the clad collapse time, which is the time required for an unsupported cladding tube to flatten into the axial gap volume caused by fuel densification. On the basis of independent staff calculations and from experience of fuel performance in other reactors, the staff concurred with the applicant that clad collapse is not expected for the ANO-1 fuel during the first cycle of 11,000 effective full power hours However, the staff concluded that the evaluation model for collapse time calculations contains several deficiencies in its application to ANO-1. The staff informed the applicant[8] that an acceptable model for collapse time calculations is necessary for subsequent fuel cycles of ANO-1.

2.3 Effects of Densification on Steady State and Transient Operation

2.3.1 General

Fuel densification can affect the steady state operation because of axial gaps in the fuel column that results in local neutron flux spikes and an overall increased linear heat rate. An additional effect occurs in the transient analyses since, due to a lower gap conductance, the fuel has a higher initial stored energy and a slower heat release rate during the transient.

On the basis of evaluations of the effects of fuel densification the ANO-1 reactor will be operated with more restrictive limits on control rod patterns and position than originally proposed, and with a reduced maximum linear heat generation rate. The limits are based on consideration of the effects of local peaking caused by gaps in the fuel pellet stack and changes in the gross peaking factors, primarily axial, which can be achieved by more restrictive operation of control rods.

The effects of densification on power density distributions have been calculated using models in conformance with those discussed in Section 4 of the staff densification report.[1] The primary calculations used the models and numerical data of the Westinghouse power spike model as described in Appendix E of that report, except that the initial nominal density used was [ ]* (the minimum density of the three batches), and the probability of gap size was changed to conform to that recommended by the staff.[1]

The calculations by the applicant take into account the peaking due to a given gap, the probability distribution of the peaks due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. The calculations result in a power spike factor that varies almost linearly with core height and reaches a maximum value of 1.15 at the top of the core. The overall calculation falls within the range examined[9,10] by our consultant, Brookhaven National Laboratory, in conjunction with reviews of other models.

*[ ] Brackets denote data known by the staff and considered proprietary to the applicant and specified in reference 6 to this report.

A normalized shape for the power spike factor is derived from power spikes caused by different gap sizes at various axial locations. The normalized shape is then used in conjunction with various axial power shapes to determine the axial location at which the decrease in DNBR due to the superimposed power spike is maximized. These calculations also include the increase in average linear heat generation rate from 5.656 kW/ft to 5.774 kW/ft due to the reduced fuel column height based on the instantaneous densification from the minimum initial density of [ ] theoretical density (TD) to a final density of 0.965 TD.[1] The reactor operating limits, which will be part of the Technical Specifications for ANO-1, are based on maximum linear heat generation rate through the reactor power vs axial offset correlation.

2.3.2 Fuel Rod Thermal Analysis

The applicant uses the B&W computer code, TAFY[11], to calculate gap conductance, fuel temperature, and stored energy for ANO-1 fuel, which in turn are used in the safety analyses. To demonstrate the applicability of the TAFY code for the evaluation of the ANO-1 fuel thermal behavior, the applicant compared TAFY predicted fuel temperatures and gap conductance with experimental data.

The staff reviewed the TAFY code and concludes that realistic or conservative assumptions have been used for modeling of the physical phenomena incorporated into the code (thermal expansion, fuel swelling, sorbed gas release, fission gas release), with two exceptions: (1) partial contact between the clad and fuel and (2) formation of a central void due to fuel restructuring on the basis of columnar grain growth at a temperature of 3200°F. Details of the staff's evaluation of the TAFY code and its application to ANO-1 type fuel rods are given in Reference 7.

Because of the two exceptions noted above, the staff required the applicant to analyse the fuel thermal performance using a 25% reduction in gap conductance and taking no credit for fuel restructuring. This analysis[(6)] resulted in a reduction in the peak linear heat rate at which centerline fuel melting would occur from 22.2 kW/ft before densification to 20.1 kW/ft after densification was conservatively taken into account. The reactor protection system prevents fuel centerline melting from occurring for all anticipated transients. This is accomplished by proper setting of the reactor trips as a function of power level and axial power imbalance. These settings will be given in the Technical Specifications.

2.3.3 Steady State and Loss-of-Flow Transient

The effect of fuel densification on the departure from nucleate boiling ratio (DNBR) during steady state operation was analyzed by both the applicant and the staff. The staff's independent calculations are described in Reference 7. The results show that the steady state minimum DNBR decreases due to an increase in the surface heat flux resulting from fuel densification. To assess the amount of reduction in DNBR margin, the applicant reanalyzed the steady state operating and design overpower conditions with an assumed axial power shape that peaked near the core outlet rather than with the symmetrical reference design power shape described in the FSAR. The outlet shape, though not achievable in operation, produces the largest possible DNBR penalty from fuel densification, because the point of minimum DNBR is shifted toward the top of the hot fuel rod where the densification induced power spike is the largest. The application of this large power spike at the point of minimum DNBR produces the greatest degradation in DNBR. Using this outlet axial power peak the applicant computed a 5.63% reduction in DNBR from the 1.55 value reported in the FSAR without the effects of densification. The applicant has proposed more stringent control rod positions and offset limits to compensate for the loss in DNBR margin. This is acceptable to the staff.

B&W also reanalyzed the loss-of flow transient that would result from a loss of electrical power to the reactor coolant pumps taking into account the effects of fuel densification. The results show that the minimum DNBR during the transient decreased due to local flux increases caused by fuel densification. The previously calculated minimum DNBR during the transient was 1.60 whereas with the densification the minimum DNBR is calculated to be about 1.53.

The densification effects that could aggravate the consequences of the loss-of-flow transient are the increase in the steady state fuel temperature (stored energy), increase in heat flux, and a decrease in gap conductance. The increase in fuel temperature provides more stored heat in the fuel which must be removed during the transient; the higher heat flux provides greater initial enthalpy in the coolant channel. The decrease in gap conductance delays the removal of heat from the fuel resulting in a higher ratio of heat flux to channel flow during the transient and thus a lower DNBR.

2.3.4 Other Transients

The following other transients have been reviewed to determine whether the effects of densification have resulted in significant changes in their consequences:

(1) Control Rod Withdrawal Incident

(2) Moderator Dilution Incident

(3) Control Rod Drop Incident

(4) Startup of an Inactive Reactor Coolant Loop

(5) Loss of Electrical Power

In the applicant's FSAR these transients were calculated to result in a DNBR in excess of 1.3, or their consequences were shown to be limited to acceptable values by limits to be set forth in the Technical

Specifications. The staff has reviewed these transients taking into account the effects of fuel densification and agrees with the applicant that they would not result in a reduction of the core thermal margin, i.e., a DNBR less than 1.3.

2.3.5 Summary

The effects of fuel densification on steady state and transient operation have been evaluated by the applicant and reviewed by the staff.

The effect on steady state operation, mostlv due to local increases in thermal neutron flux and heat generation, is to require more restrictive limits on control rod positions and offset limits in the Technical Specifications for ANO-1. In order to prevent fuel melting the maximum allowable linear heat generation rate has been reduced from 22.2 kW/ft to 20.1 kW/ft. The overpower trip limit has been reduced from 114 percent to 112 percent such that a DNBR greater than 1.3 is maintained for steady state and during transient conditions.

The staff concluded on the basis of its review that the potential effects of fuel densification on steady state and transient operation have been evaluated in an appropriate manner and are acceptable for the period of operation proposed.

2.4 Accident Analyses

2.4.1 General

Analyses of the consequences of various postulated accidents were presented in the FSAR for ANO-1. The accidents evaluated were:

(1) Locked Rotor

(2) Loss-of-Coolant (LOCA)

(3) Control Rod Ejection

(4) Steam Line Rupture

(5) Steam Generator Tube Rupture

(6) Fuel Handling

(7) Waste Gas Tank Rupture

Since fuel densification will affect the consequences of the first four postulated accidents they have been reanalyzed by the applicant and reevaluated by the staff. Results of the first three accidents (locked rotor, loss-of-coolant, and control rod ejection) are presented in separate parts of this section. The steam generator tube rupture, waste gas tank rupture, fuel handling and steam line rupture accidents are discussed below.

Changes in the fuel pellet geometry can cause the stored energy in the fuel pellet to increase by the mechanisms discussed in Section 2.3 of this report. Potential increases in local power due to the formation of axial gaps are discussed in Section 2.3.1. Both of these effects are accounted for in the evaluation of accidents.

The radiological consequences of accidents were independently calculated by the staff. The results of the staff's calculation of the radiological consequences of accidents were presented in the ANO-1 Safety Evaluation Report dated June 6, 1973. The radiological consequences would not increase as a result of fuel densification, although the transient performance of the fuel rods can change as a result of fuel densification. It is the latter factor that is discussed in the following sections.

The staff evaluation of the radiological consequences of a waste gas decay tank failure was based on an assumed quantity of gas in the tank limited by the Technical Specification. For the steam generator tube rupture accident, the assumed quantity of reactor coolant activity is consistent with the Technical Specification limits on maximum permitted reactor coolant system activity. Fuel densification will not affect the consequences of these accidents.

The postulated refueling accident assumes the dropping of a fuel assembly in the spent fuel pool or transfer canal. The fuel rods are assumed to be approximately at ambient temperature during the postulated accident. Therefore, the direct effects of fuel densification will not affect the consequences of this postulated accident. The potential for mechanical failure of a flattened rod might be different from that of a normal rod; however, since the staff evaluation has been based on the conclusion that no clad collapse will occur during the fuel cycle (Section 2.2), this potential change in fuel rod characteristics was not considered. Furthermore, all of the rods in the dropped assembly are assumed to fail.

The steam line break accident was analyzed by the applicant in the FSAR without the effects of fuel densification. That analysis showed that the worst consequences from this accident would result at the end of life (EOL) of the core. Since the DNBR margin is higher at the EOL, including the effects of fuel densification, the staff does not expect that the thermal limits will be more severe than those presented in the FSAR.

2.4.2 Locked Rotor Accident

The reactor coolant system for ANO-1 consists of two loops; each return from the steam-generator to the reactor consists of two cold legs, i.e., a total of four reactor coolant pumps are used. Locked rotor accidents are characteristically less severe for 4 pump plants than for 3 or 2 pump plants.

The analysis of the locked rotor accident was originally presented in Section 14 of the FSAR. The transient behavior was analyzed by postulating an instantaneous seizure of one reactor pump rotor.

The reactor flow would decrease rapidly and a reactor trip would occur as a result of a high power-to-flow signal. The core flow would reduce to about three fourths its normal full-flow value within two seconds. The temperature of the reactor coolant would increase, causing fluid expansion with a resultant pressure transient which would reach a peak of approximately 15 psi above nominal. The applicant computed a maximum cladding temperature of 1350°F at about 4.5 seconds for this accident.

The staff performed independent calculations for this postulated accident using Oconee Unit 1 parameters. The results of these calculations, which are discussed further in Reference 7, showed that calculated cladding temperatures varied from 670°F to 1720°F, all acceptably low, even with conservative variation of input parameters.

2.4.3 LOCA Analysis

The B&W evaluation model described in the AEC Interim Acceptance Criteria and Amendments for Emergency Core Cooling Systems was used by the applicant to evaluate the loss-of-coolant accident (LOCA) for ANO-1. The analysis was performed with the B&W CRAFT code for the blowdown period and the THETA code for the fuel rod heat up. The applicant's LOCA analysis without the assumption of fuel densification is reported in the FSAR based on the 8.55 ft^2 split break in the cold leg at the pump discharge as the limiting break size and location.[12]

During the blowdown period the gap conductance, reduced due to fuel densification according to the staff requirements, could cause the core average fuel pellet temperature to increase, but CRAFT calculations show that the temperature experiences only a very small change. Since in the initial analysis an average core temperature was used that is higher than the average core temperature resulting from the decreased gap conductance, the applicant concludes that the limiting break size and locations do not change due to fuel densification.

The effects of fuel densification on the reflood calculation is small, since the gap conductance is much larger than the film coefficient (cladding surface to coolant) during reflood. The film coefficient is thus limiting with regard to heat transfer and cladding temperature.

The applicant performed the LOCA analysis with an axial power shape that peaks [ ] below the core midplane and a corresponding axial peaking factor of $F^Z = 1.816$ which includes an axial uncertainty factor of 1.024 and a local factor of 1.026 accounting for the effect of the grid structure on the axial peak. This particular flux shape results in the highest linear heat rate and occurs during the control rod maneuvering resulting from the 4-day design basis transient. The design basis transient is defined as a 100% -30% -100% transient, consisting of operation at 100% power, reduction to 30% power, operation at 30% power for about 8 hours, and return to 100% power.

The THETA calculations were performed with the staff requirements for initial fuel pellet density assumptions. However, instead of imposing a power spike due to a fuel column gap at the peak axial power [ ] below the core midplane the applicant used an equivalent radial multiplier over the entire length of the fuel pin which leads to a higher calculated peak cladding temperature of approximately 10°F. A hot channel factor of F_{HC} = 1.014 was used in the calculations. The radial peaking factor, F^R, including an uncertainty factor of 1.05 was varied until the calculated maximum cladding temperature approached the 2300°F limit. Using the gap conductance as calculated with the TAFY code described in Section 2.3.2 a clad temperature of 2283°F was reached with a maximum linear heat rate of 18.5 kW/ft, which, therefore, is the maximum allowable linear heat generation rate for the ANO-1 reactor.(6) In order to accommodate a possible quandrant tilt of 5% during this design basis transient the allowable heat rate is further reduced to 16.65 kW/ft. The maximum allowable linear heat rate will be controlled by a control rod operating band.

2.4.4 Rod Ejection Accident

The control rod ejection transient has been reanalyzed(4,5) by the applicant to account for changes in the fuel due to densification. The significant effects of fuel densification are an increase in the initial maximum fuel temperature and a slight increase in average heat flux due to shrinkage of the pellet stack length. In addition, spikes in the neutron power can occur due to gaps in the fuel. Calculations have verified that no changes in the basic kinetic response of the core occur due to the small changes in fuel geometry and heat transfer characteristics.

The results of the rod ejection accident at BOL and at EOL without consideration of densification effects have been previously presented in the FSAR. The staff consultants at Brookhaven National Laboratory (BNL) have performed independent check calculations using appropriate input data and their own computer codes and have confirmed that the results of a rod ejection transient are less severe at EOL than at BOL. Therefore, all calculations by the applicant considering densification effects were done for BOL conditions.

For the full power transient, the control rod reactivity worths available for the assumed ejected rod would be expected to decrease because of the more restrictive insertion limits on the control bank. However, this was not included in the reevaluation, thereby adding additional conservatism to the calculations. The maximum Technical Specification rod worth of 0.65% delta k/k was used for the BOL calculations at full power.

The staff review of the initial fuel temperature for the BOL full power case indicated that a reasonable temperature was used for the assumed conditions, consistent with that used in the LOCA analysis. The neutron power spike effect was included in the reanalysis.

The reexamination of the rod ejection transient considering the effects of densification has resulted in a peak pellet average enthalpy below the staff's criterion of 280 cal/gm, and the maximum clad temperature during the transient is 1510°F. The fraction of fuel pins calculated to be in DNB is 28%. The staff review of the rod ejection analysis indicates that reasonably conservative consideration has been given to the effects of fuel densification and that the results are acceptable for this accident.

2.5 Classification and Selective Loading of Fuel

2.5.1 Background

The fuel densification report issued by B&W for the ANO-1 core[6] was concerned with Batch 1 fuel, which was manufactured to a nominal density of 92.5% T.D. and a diameter of 0.370 in. This fuel was loaded without resintering. Batches 2 and 3, however, while initially manufactured to the above specifications, were subsequently resintered to a higher nominal density. The resintering of the already manufactured fuel resulted in variations in pellet diameter and density. For some of these pellets the diameter and density variations were large enough to effectively reduce the allowable heat rate to which they could be exposed during reactor service.

To ensure that these heat rate limited pellets would not be used in core positions where power peaks might occur that would result in these exceeding the allowable levels, the applicant classified all Batch 2 and 3 fuel assemblies according to the maximum allowable heat rate at which they can be separated. These fuel assembly classifications were used, in conjunction with existing fuel cycles calculations, to establish the ANO-1 loading plan for the first cycle, and the two subsequent cycles.

The applicant defined three classifications into which the ANO-1 fuel was divided. Fuel assemblies that could be operated at the allowable peak linear heat generation rate (LHGR) were called Class 1. The remaining classes were equally separated at intervals of three percent of the maximum linear heat rate, i.e., a Class 3 fuel assembly would be limited to operation at a peak LHGR no greater than 94 percent of the peak core design LHGR.

Classification of fuel assemblies was performed as follows:

1. The dimensions of the resintered, pellets were determined for each pellet lot by use of a statistical sampling plan.
2. The "as-built" fuel pellet data were analyzed to determine the maximum allowable peak LHGR as a function of density and diameter, and the pellets were classified accordingly.
3. The fuel lots were then loaded into fuel rods, and the rods were classified to correspond to the classification of the fuel with which they were loaded. The rods were in turn loaded into fuel assemblies and the assemblies classified to correspond to the lowest classification of any fuel rod in that assembly.

2.5.2 Analyses

The Regulatory staff has reviewed the applicant's assumptions, methods, and computer codes used in evaluating the effects of classification and selective loading of fuel. The staff has also examined the applicant's procedures for verification of fuel loading. The staff agrees with the applicant that from a relative standpoint the limiting criterion for the classification of fuel assemblies is fuel centerline melting in design thermal transients as opposed to the LHGR for loss of coolant accidents.

The analytical model used by the applicant in determining the limiting LHGR's was the TAFY computer code. The analysis was conducted by the staff approved version of the code. The staff concurs with the applicant that if at a constant density the diameter is reduced, then the maximum allowable LHGR to prevent fuel melting is also decreased, and that if at a constant diameter the initial density is reduced, then the maximum allowable LHGR to prevent fuel melting is similarly reduced. This result ensues from an increase in the fuel-clad gap with an accompanying decrease in the heat transfer across that gap.

The applicant's sampling plan as it applied to fuel pellet density and diameter has been reviewed. The sampling plan determines the mean and lower tolerance limit on diameter and density and thus is a part of the basis for classifying the fuel pellets. When classifying pellet groups based on statistical sampling results, the lower tolerance limits (LTL) on density and diameter were used. The reason for utilizing the LTL as an acceptance criterion was to ensure that the tolerance ranges on groups of pellets remained within the original specified criterion. The original range on diameter was ± 0.0005 inch and the original range on percent theoretical density was ± 1.5%. There is assurance, therefore, that if a group of pellets falls within Class 2 or 3 acceptance criteria the distributions associated with the density and diameter are within acceptance limits.

The staff concludes on the basis of its review that the potential effects of the applicant's classification and selective loading techniques on the resultant effects of fuel densification on steady state and transient operation have been evaluated in an appropriate manner and are acceptable.

2.5 Summary and Conclusions

The effects of fuel densification have been considered in analyses of normal operation, operation during transient conditions, and postulated accident conditions. On the basis of the staff review of the applicant's calculations, and independent calculations performed by the staff and its consultants, the staff concluded that for the period of operation proposed, namely the first fuel cycle:

(1) The effects of densification during steady state and transient operation of the ANO-1 reactor will not cause the limits on DNBR, cladding strain, and centerline temperatures, to become less conservative than values previously established in the FSAR.

(2) The effects of densification were included in the calculation of fuel rod behavior during postulated loss-of-coolant accidents. The LOCA analysis is acceptable and complies with the June 1971 Interim Acceptance Criteria.

(3) The applicant's omission of the creep down effect, which tends to increase gap conductance with life time, is acceptable.

(4) The Technical Specifications will limit the fuel residence time

to 11,000 effective full power hours of power operation to assure no cladding collapse.

(5) The appli- ınt has adopted the staff recommendations for calculating gap cond- and fuel temperatures (Section 2.3.2) as they are used - state, transient and accident conditions.

(6) Operating restrictions as necessary to assure compliance with items (1) through (4) above will be incorporated into the Technical Specifications.

On the basis of the above summary, the staff concludes that the applicant is in compliance with the staff densification report[1] and that ANO-1 can be operated at power levels up to 100% of rated power with no undue risk to the health and safety of the public.

3.0 Conclusion of Remaining Review Items

When the staff Safety Evaluation Report was published on June 6, 1973 a number of safety concerns were not yet fully resolved by the applicant. These items were identified in the Safety Evaluation Report, principally in Sections 7.0 and 8.0. The resolution of these items is discussed in the following section along with a new item, the pump backstop failure which was identified during the ANO-1 test program.

3.1 Emergency Feedwater System

If the ANO-1 reactor is shut down, decay heat must be removed from the reactor through at least one of the steam generators until the plant has been cooled and depressurized sufficiently to permit use of the Decay Heat Removal System. As long as off-site power is available, the Main Feedwater and Condensate Systems are able to furnish the needed flow to the steam generator. However, to provide an assured source of feedwater if off-site power is lost, the applicant has provided the Emergency Feedwater System (EFWS). The EFWS consists of two full capacity EFW pumps, two sources of feedwater and the piping, valves, and controls needed to deliver feedwater to either or both steam generators. The decay heat removal is achieved by delivering feedwater to an intact steam generator and venting the generated steam through the turbine bypass valves, the atmospheric dump valves, or the steam relief valves.

The applicant was reminded that the EFW system is required for safety and as such it should meet the single failure criterion; and the instrumentation, control and electrical equipment should be designed to conform with IEEE-279 and IEEE-308. The applicant amended the design to satisfy the above stated criterion and standards with but one exception, and has provided the results of a single failure analysis and design modifications of the EFW system in Amendment 42 of the FSAR.

The final design of the EFW system includes two redundant full capacity pumps. The piping is arranged to permit either pump to deliver emergency feedwater to either one of the two steam generators. One pump is driven by a steam turbine receiving steam from either one of the steam generators. The other pump is driven by an electrical motor which is normally powered from one of the main 4160 V non-emergency buses (bus A1). Upon loss of offsite power, the EFW motor-driven pump can be supplied from one of the diesel generators by manually connecting bus A1 to emergency bus A3. An analysis has been performed showing that the diesel generator is capable of withstanding this additional load without the need of shedding other connected safety loads and without infringing upon the recommendations set forth in Regulatory Guide 1.9. Both manual and automatic controls are provided to establish EFW flow paths to the steam generators. The automatic function is accomplished through the Integrated Control System (ICS). In view of the non-safety grade status of the ICS, an analysis has been performed by the applicant to establish that in the event of a failure in the automatic control system there is sufficient time for the operator to manually initiate the operation of the EFW system before the core is endangered.

We have reviewed the instrumentation, control and electrical aspects of the final design of the EFW system and have concluded that the design would satisfy the single failure criterion, IEE-279 and IEEE-308 and be acceptable with the satisfactory resolution of the following items:

1. With regard to the EFW motor-operated supply valve identified as CV2620 in Figure 14.11-1 of Amendment 42 of the FSAR, the valve motor must be connected to Bus A4 to meet the single failure criterion. The applicant has agreed to this change.
2. With regard to ICS control of the EFW system, the applicant has proposed to install Class 1E isolation devices to prevent failures in the ICS from propagating to the EFW system. We have not reviewed the details of the proposed isolation scheme, and until we have reviewed and accepted it do not consider the EFW system acceptable with automatic control by the ICS. We do consider the EFW system acceptable if the ICS is disconnected and the EFW system is operated manually. We will require the ICS control to be disconnected from the EFW system unless the applicant can demonstrate acceptable design changes which will prevent failures in the non-Class 1E system from affecting the EFW system.
3. The power supply for the motor-driven pump, bus A1, is located in the Turbine Building and is not Class 1E switchgear. The applicant has evaluated the A1 bus with regard to floods and seismic events.

Bu Al is installed at the 372-feet elevation which is high enough to be protected from the Probable Maximum Flood. With regard to seismic events, the Al switchgear is essentially similar to the Class 1E units and is installed in a heavily constructed portion of the Turbine Building with the deck above and surrounding switchgear providing inherent protection from missiles or falling objects. No high energy fluid piping is routed in this area; nor are there any items of mechanical equipment or other potential hazards located near Bus Al.

The power and control cables for the motor-driven EFW pump are not routed through engineered safguard raceways. However, they are installed in accordance with the same procedures and the power cables themselves were purchased under the terms of the same specification as the safeguards cables.

In view of the foregoing considerations, we conclude that the use of Bus Al for the power supply to the motor-driven EFW pump is acceptable.

Therefore, with the resolutions indicated above, we find the ANO-1 EFW system acceptable.

3.2 Steam Line Break Isolation

The applicant has analyzed the response of the ANO-1 reactor to the uncontrolled blowdown of a single steam generator caused by a postulated steam line break. As presented in Section 14.2.2 of the FSAR, the analysis shows a return to 2.6% power after a blowdown of one steam generator, occurring in less than one minute. The analysis of a blowdown of both steam generators was requested since the main steam block valves were set up for remote manual control and the main feedwater valves are closed by the non-safety grade ICS. The applicant committed to install a reliable system of isolating the seismic Category I sections of the system to preclude such double blowdown. This system, the steam line break instrumentation and control system (SLBIC), will sense low steam pressure and automatically close the main steam and main feedwater block valves. We reviewed the original design of the SLBIC and found it unacceptable since it did not meet all the requirements of IEEE-279. The applicant has since revised the design of the SLBIC and resubmitted it for our review. The review of the revised SLBIC design is underway and installation of the system is expected to be completed by late summer. Since the SLBIC protection is needed only later in core life when the moderator temperature coefficient has changed from the initial positive value to a significant negative value, and the protection of the ICS is available in the interim, we consider ANO-1 acceptable for licensing at this time with respect to steam line break isolation.

3.3 Offsite Power Connections

Section 8.3 of the SER noted our concerns with regard to indiscriminate tripping of available offsite power supplies. Also, potential single failures were identified which could result in the loss of both offsite and onsite power to the emergency buses.

We have reviewed the applicant's design modifications and have concluded that the modified offsite power system and ac emergency onsite power system meet our requirements, and they are acceptable.

3.4 Cable Arrangement In Control Room And Rod Drive Control (RDC) Equipment Room

Section 7.9 of the SER reflected our concerns with regard to cable arrangements in (1) control room subfloor, (2) RDC equipment room subfloor, and (3) control room overhead. The applicant has either demonstrated the adequacy of the cable arrangement design or modified the design to make it acceptable as follows:

3.4.1 Control Room Subfloor Cable Arrangement

Lack of cable separation and vulnerability to common mode failures resulting from design basis events such as fire and flooding were our concerns with regard to the design arrangement of redundant RPS cables in the control room subfloor. The applicant has indicated that these cables carry only low-energy signals and have waterproof hypalon or neoprene jackets. The cables are enclosed in flexible sealtite conduits which have a polyvinyl chloride (PVC) covering. The applicant claims that the PVC material is fire retardant and self-extinguishing. Additionally, steel spacers or a board of fire resistant material (Industrial Marinite) are used to separate conduits of redundant channels to eliminate the possibility of a fire at one conduit from igniting an adjacent one. Also, a Halon fire suppression system, designed to meet or exceed NFPA (National Fire Protection Association) requirements, is installed in the control room subfloor. This system is activated automatically at 160°F by heat detectors installed throughout the subfloor area, or can be activated manually. Smoke detectors are also installed in the subfloor area to actuate alarms in the control room. CO_2 fire extinguishers are available nearby for backup protection.

We have reviewed the proposed design modifications and have concluded that the provisions of the design to minimize the probability and the effect of design basis events from rendering RPS cables inoperable are acceptable.

3.4.2 RDC Equipment Room Subfloor Cable Arrangement

The cable design arrangement in the RDC equipment room subfloor was of concern for the same reasons stated before for the RPS cables. The applicant has indicated that all safety related cables are routed in rigid steel conduits and there is no PVC material in this area. We have determined that this is acceptable.

3.4.3 Control Room Overhead Cable Arrangement

Our concern was the open raceways containing RDC power cables located overhead in the control room. We concluded that these power cables were a potential source of fire that could affect the availability of both Unit 1 and Unit 2 control rooms. We required that the applicant install a fire barrier separating these open raceways from the control room proper, and provide the accessibility and means necessary to extinguish a fire quickly.

The applicant has complied with our position and, in addition, has installed a Halon fire suppression system. We have concluded that this is acceptable.

3.5 Reactor Coolant Pump Backstop Failure

In a letter report dated November 19, 1973 the applicant notified the AEC regional office of the results of the investigation of the reactor coolant pump backstop failure which occurred during the preoperational test program. The failure occurred when one of the operating reactor coolant pumps was shut down. When it coasted to a stop the backstop, the anti-reversing device, failed to engage and the hydraulic forces generated by the operating pumps caused this one pump to start rotating in reverse. After some reverse speed was attained the backstop abruptly engaged but was damaged by absorbing the inertial force of the spinning 16-ton rotor. Upon disassembly, parts of the backstop were found to be deformed and the key anchoring the backstop to the top of the motor frame was sheared. Inspection of the other backstops in ANO-1 revealed evidence of a less severe but similar failure on one of the other pumps.

The applicant's investigation showed that the damage was confined to the backstop assembly and related parts. The applicant concluded that the failure of the backstops to engage was due to inadequate part clearances. The backstop assemblies have been repaired and returned to service. The applicant concluded that the backstop failure is not safety related.

The staff has evaluated the backstop failure and concluded that the applicant's corrective actions were appropriate. However, the performance of the backstop may be significant to safety when considering the possibility that blowdown through a reactor coolant pump after a loss-of-coolant accident may drive the pump rotor to excessive speeds. That concern is being considered generically (see Section 4.3 of this Supplement). The implications of the backstop failure are being considered in that context. Aside from that ANO-1 is considered acceptable because the mechanical integrity of the reactor coolant system has not been compromised and the backstop function after normal pump shutdown is not a safety concern.

4.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS or Committee) reviewed the ANO-1 application at its August 1973 meeting and subsequently, reported its findings to the Commission by letter dated August 14, 1973. The ACRS letter is attached as Appendix B. The following sections describe AEC Regulatory staff actions with respect to specific issues that were identified in the ACRS report.

4.1 Operation at 2568 MWt

The Committee stated that the operation of the prototype, Oconee Unit 1, at power levels up to 2452 MWt should be satisfactory to the staff before ANO-1 is operated a full license power (2568 MWt). The staff made that determination based on performance of Oconee Unit 1 at power levels up to 2452 MWt. That performance is described in the Oconee startup report[14].

4.2 Positive Moderator Temperature Coefficient

The Committee's concern with regard to operation with a positive moderator temperature coefficient will be resolved in a manner satisfactory to the staff in the Technical Specifications. The moderator temperature coefficient will be restricted to values less (more negative) than those values employed in the safety evaluation accident analysis. The Technical Specifications will prohibit operation above 95% power unless the moderator temperature coefficient is zero or negative since those were the conditions of the LOCA safety analysis.

4.3 Pump Overspeed

The staff is investigating on a generic basis the consequences of an unlikely rupture of a reactor coolant pipe which in certain locations might result in reactor coolant pump overspeed. If this study indicates that additional protective measures are warranted to prevent significant pump overspeed or the potential consequences to safety related equipment, the staff will require the applicant to provide these protective measures.

4.4 Common Mode Failure and Anticipated Transients Without Scram

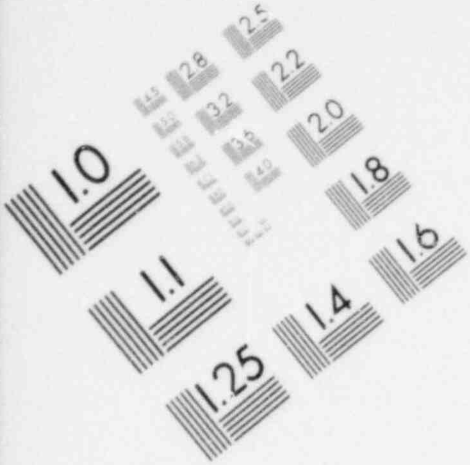
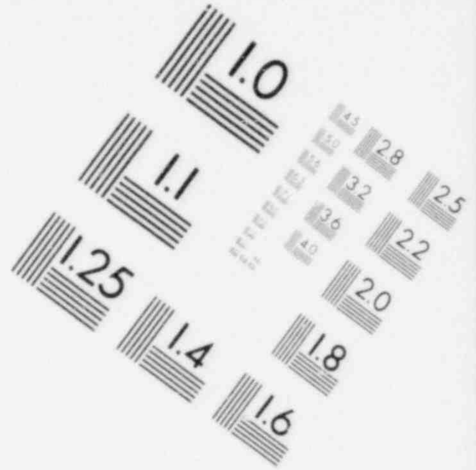
The staff's position with regard to this potential problem is stated in an October 9, 1973 letter to the applicant calling for detailed analysis and design modifications, if necessary.

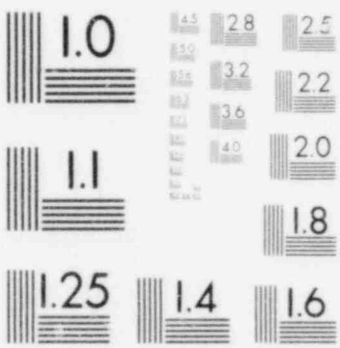
4.5 Control of Power Peaking Factors and Linear Heat Rate

The Committee recommended that the staff establish suitable criteria for those measures which will be taken to prevent operating under conditions which might result in exceeding acceptable fuel limits established from accident studies and other considerations.

The applicant is providing alarms and administrative procedures acceptable to the staff to prevent exceeding acceptable fuel limits. In addition, power distribution maps will be required periodically during steady state and following transient operation in order to verify predicted power distributions.

IMAGE EVALUATION
TEST TARGET (MT-3)



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MICROCOPY RESOLUTION TEST CHART

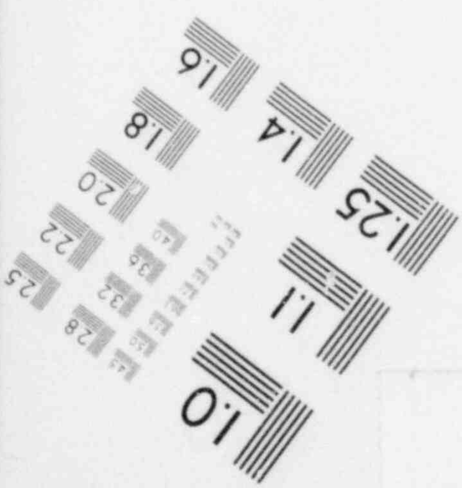
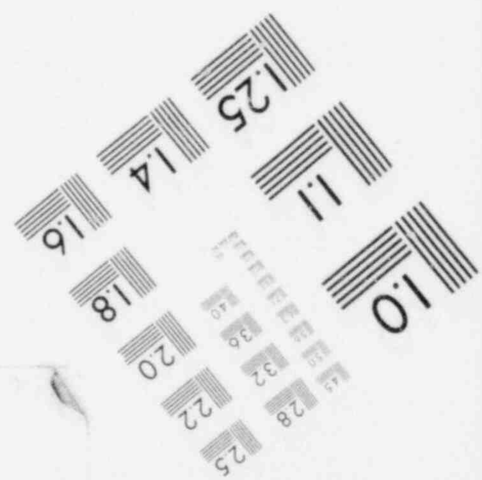

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

4.6 Changes in AEC ECCS Acceptance Criteria

The AEC ECCS Acceptance Criteria have been revised and were published on December 28, 1973. The ANO-1 operating limits will be re-evaluated and changes acceptable to the sta.f will be incorporated into the Technical Specifications.

4.7 Course-of-Accident Instrumentation

The ACRS suggested that the applicant assure itself that instrumentation for determining the course of potentially serious accidents, on a time scale that will permit appropriate emergency action, is provided at the station. In addition, the applicant was asked to assure that appropriate calibration methods and calculated bases for interpreting instrument responses are available. The applicant stated assurance with regard to these matters in a letter to the staff dated September 10, 1973.

The Regulatory staff, in the operating license review, found ANO-1 acceptable with respect to course-of-accident instrumentation for the following reasons:

1. Safety related instrumentation in the reactor building is qualified to operate in the post accident environment.
2. The status of engineered safety features is displayed in the control room so that the operator can verify system operation or take prompt corrective action where necessary.

3. The calculated responses to major accidents as presented in the Final Safety Analysis Report enable a trained operator to judge the adequacy of system response after an accident.

4.8 Safety Review Committee

The Committee recommended that the applicant's Safety Review Committee include additional experienced personnel from outside the AP&L corporate structure as voting members. The applicant has subsequently proposed to amend the Technical Specifications for ANO-1 to require a Radiation and Health Physics Consultant and a Nuclear Safety Consultant as voting members of the corporate Safety Review Committee. The Regulatory staff considers this requirement acceptable and will include it in the Technical Specifications issued for ANO-1.

5.0 CONCLUSION

The staff's conclusions as stated in the SER remain unchanged.

6.0 REFERENCES

1. "Technical Report on Fuel Densification of Light Water Reactor Fuels," Regulatory Staff, U.S. Atomic Energy Commission, November 14, 1972.
2. "Fuel Densification Report," BAW 10054 Topical Report (Proprietary), January 1973 (Nonproprietary Information in BAW 10055).
3. "Oconee 1 Fuel Densification Report," BAW 1387 (Proprietary), January 1973 (Nonproprietary Information in BAW 1388).
4. "Fuel Densification Report," BAW 10054 - Rev. 1 Topical Report (Proprietary), April 1973.
5. "Oconee 1 Fuel Densification Report," BAW 1387 - Rev. 1 (Proprietary), April 1973.
6. "Arkansas Nuclear One Unit 1 Fuel Densification Report," BAW 1391 (Proprietary), June 1973. (Nonproprietary information in BAW 1392.)
7. "Technical Report on Densification of Babcock & Wilcox Reactor Fuels" by the Regulatory Staff, U.S. Atomic Energy Commission, July 6, 1973.
8. Letter from R.C. DeYoung to R. Edwards, Babcock & Wilcox dated April 23, 1973, with copy to Arkansas Power & Light Company.
9. "Peaking Factors in Pressurized Water Reactors with Fuel Densification," BNL Interim Report, December 1972.
10. "Peaking Due to Densification in the Maine Yankee Reactor;" BNL Interim Report, March 1973.
11. "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," BAW 10044, Topical Report, April 1972.

12. "Multinode Analysis of B&W's 2568-MWT Nuclear Plants During a Loss-of-Coolant Accident," BAW 10034, Rev. 1, May 1972.

13. Letter from Duke Power Company to A. Giambusso, dated May 14, 1973.

14. Duke Power Company Oconee Nuclear Station Unit 1 (Docket No. 50-269), License No. DPR-38, Startup Report dated November 16, 1973.

Appendix A

<u>Supplement to the Chronology of the</u>

<u>Regulatory Staff's Operating License</u>

<u>Review of Arkansas Nuclear One, Unit 1</u>

June 15, 1973	Applicant filed Amendment # 38 to the FSAR.
June 15, 1973	Applicant filed BAW-1391, Arkansas Nuclear One Unit 1 Fuel Densification Report.
June 16, 1973	Meeting with the applicant at Bethesda to discuss outstanding commitments.
June 20, 1973	Applicant filed Amendment #39 to the FSAR.
July 26, 1973	Advisory Committee on Reactor Safeguards subcommittee meeting held in Washington, D.C.
August 6, 1973	Applicant filed Amendment #40 to the FSAR.
August 9, 1973	Advisory Committee on Reactor Safeguards full committee meeting held in Washington, D.C.
September 13, 1973	Additional information requested of applicant on special loading of re-sintered fuel.

September 24, 1973	Applicant filed Amendment #41 to the FSAR.
October 9, 1973	AEC letter to applicant on Anticipated Transients Without Scram (ATWS)
October 24, 1973	AEC visited site for final review of high energy line rupture outside containment.
October 24, 1973	Applicant requested extension of Construction Permit from January 1, 1974 to May 1, 1974.
October 30, 1973	Meeting with applicant at Bethesda on outstanding electrical review items.
November 12, 1973	Applicant filed proprietary report on analysis and selective loading of resintered fuel.
November 16, 1973	Letter to applicant describing outstanding electrical review concerns.
November 30, 1973	Applicant filed Amendment #42 to the FSAR.
November 30, 1973	Applicant filed supplementary information by letter on electrical review items.

December 26, 1973	Letter to applicant on increased surveillance of high energy lines.
January 15, 1973	Applicant letter to AEC on resolution of outstanding electrical ıtems.

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

August 14, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ARKANSAS NUCLEAR ONE-UNIT 1

Dear Dr. Ray:

During its 160th meeting, August 9-11, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application of the Arkansas Power and Light Company for a license to operate Arkansas Nuclear One-Unit 1 (formerly Russellville Nuclear Unit) at power levels up to 2568 MW(t). The site was visited by a Subcommittee on May 4, 1973, and the project considered during a Subcommittee meeting held in Washington, D. C., on July 26, 1973. In the course of the review, the Committee had the benefit of discussions with representatives and consultants of the Arkansas Power and Light Company, the Babcock and Wilcox Company, the Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed. The Committee last reported to the Commission on the construction of this unit in its letter of September 12, 1968, and on Unit 2 in its letter of February 10, 1972.

Arkansas Nuclear One is located about six miles from Russellville, Arkansas, on a peninsula formed by the Dardanelle Reservoir on the Arkansas River.

The application for a construction permit proposed initial operation at power levels up to 2452 MW(t), the same as the construction permit power level of Oconee Nuclear Station Unit 1 which employs a similar reactor. Safety studies and performance analyses have been made for a power level of 2568 MW(t) for Arkansas Nuclear One-Unit 1. The Committee believes that review of the operation of Oconee Nuclear Station Unit 1 by the Regulatory Staff should be completed and satisfactory performance of Oconee Nuclear Station Unit 1 should be demonstrated before Arkansas Nuclear One-Unit 1 is operated at full licensed power.

DR-6055

The hot functional testing of Oconee Nuclear Station Unit 1 which was conducted in 1972 caused damage of some components, including reactor vessel internals. The design changes which were made for Oconee Nuclear Station Unit 1 have been applied to Arkansas Nuclear One-Unit 1. The Committee believes that these changes are acceptable.

The applicant has been responsive to the Committee's recommendation that suitable instrumentation be sought to monitor for loose parts and for vibration; such instrumentation has been designed and will be utilized.

The applicant stated that he will propose appropriate additional operating limitations if, at any time during operation, the moderator temperature coefficient of reactivity is positive. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Regulatory Staff has been investigating on a generic basis the problems associated with a potential reactor coolant pump overspeed in the unlikely event of a particular type of rupture at certain locations in a main coolant pipe. Some additional protective measures may be warranted and this matter should be resolved to the satisfaction of the Regulatory Staff. The Committee wishes to be kept informed.

The Committee reiterates its previous comments on the need for further study of means for preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The applicant should assure himself that instrumentation for determining the course of potentially serious accidents, on a time scale that will permit appropriate emergency action, is provided at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

In view of the important role of the applicant's Safety Review Committee in providing continuing reviews, and in updating and implementing safety measures, the ACRS recommends that the Safety Review Committee include additional experienced personnel from outside the corporate structure as voting members.

The applicant has proposed measures, including alarms and administrative procedures, to prevent operating under conditions which might result in exceeding acceptable fuel limits established from accident studies and other considerations. The current review has been confined to the first fuel cycle and the analyses have been based on the as-built fuel. The ACRS recommends that the Regulatory Staff establish suitable criteria for these measures, and provide suitable bases for evaluating future loadings. The Committee wishes to be kept informed.

The Committee recognizes that re-evaluation of operating limits may be necessary as a result of possible changes in the acceptance criteria for emergency core cooling systems. The Committee wishes to be kept informed.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that Arkansas Nuclear One-Unit 1 can be operated at power levels up to 2568 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References attached.

References - Arkansas Nuclear One-Unit 1

1. Final Safety Evaluation Report, Volumes I through IV
2. Amendments [illegible] hrough 39 to the Application
3. Arkansas Power and Light Company (AP&L) letters dated October 2 and 25, 1972, transmitting lists of B&W Topical Reports for ANO-1
4. AP&L letter dated February 28, 1973, notifying AEC of its intent to incorporate the Winter 1972 Addenda of ASME Section III into the requirements of a valve purchase order for ANO-1
5. AP&L letter dated March 13, 1973, regarding requirements in electrical instrumentation and control systems at ANO-1
6. AP&L letter dated April 11, 1973, furnishing information regarding engineered safeguards control circuits
7. AP&L report dated April 1973, "Interim Report on Fuel Densification for ANO-1"
8. AP&L letter dated April 23, 1973, furnishing information on stress profiles for the main steam and main feedwater lines
9. AP&L letter dated May 11, 1973, furnishing responses to AEC requirements for electrical instrumentation and control systems
10. AP&L letter dated May 11, 1973, furnishing responses to AEC requirements to modify design of emergency cooling reservoir at ANO-1
11. DL Safety Evaluation for ANO-1, dated June 6, 1973
12. DL Technical Report on Densification of B&W Reactor Fuels, dated July 6, 1973
13. Letter from Mrs. Robert H. Douglass, Russellville, Arkansas, dated July 17, 1973, regarding ANO-1 and Subcommittee Meeting July 26, 1973

dividends, of \$65.8 million was up 57% over 1966. The pertinent financial ratios indicate an adequate financial position, and these are in line with ratios of the electric utility industry as a whole. These ratios as of December 31, 1971 are: long-term debt to net utility plant - .52; net plant to capitalization - 1.12; proprietary ratio - .35; operating ratio - .78; rate of earnings before interest on total investment - 6.3%; rate of earnings on stockholders' equity - 9.4%; times interest earned on long-term debt - 2.5; and retained earnings - \$239.6 million.