

UNITED STATES GOVERNMENT

Memorandum

TO : THE FILES
THRU: Roger S. Boyd, Chief
Research & Power Reactor Safety Branch, DRL

FROM : B. Grimes
Research & Power Reactor Safety Branch, DRL

SUBJECT: MEETING WITH DUKE POWER COMPANY ON FORTHCOMING CONSTRUCTION PERMIT APPLICATION

DATE: SEP 8 1966

50-269

On August 25, 1966, a meeting was held in the Bethesda offices with representatives of Duke Power Company, Babcock & Wilcox Company and Bechtel Corporation to discuss Duke's forthcoming application for a construction permit for the Keowee River site in North Carolina. Duke will be its own architect-engineer but has hired Bechtel as a consultant on the prestressed, post-tensioned containment design and other areas. Babcock and Wilcox will provide the nuclear steam supply system.

Attendance at the meeting included the following:

M. M. Mann	AEC-REG
E. G. Case	AEC-DRL
R. S. Boyd	AEC-DRL
D. R. Muller	AEC-DRL
B. Grimes	AEC-DRL
C. Long	AEC-DRL
R. L. Waterfield	AEC-DRL
W. C. Seidle	AEC-CO
Gene Watkins	Duke
R. L. Dick	Duke-Constr.
L. C. Dail	Duke-Engr.
E. C. Fiss	Duke-Engr.
Roy B. Snapp	Duke (Wash.)
D. S. Robbins	Duke Engr.
T. F. Wyke	Duke Engr.
W. H. Owen	Duke Engr.
P. H. Barton	Duke-Steam Production
S. E. Nabow	Duke-Steam Production
A. C. Thies	Duke-Steam Production
D. W. Montgomery	B&W - Project Manager
R. E. Wascher	B&W - Engr.
W. S. Lee	Duke VP for EWGR
C. D. Stratton	Bechtel Corporation
N. F. Rau	Bechtel Corporation



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Duke Power Company representatives stated that the present schedule called for an application for construction permit to be submitted about December 1, 1966. Future schedule dates are as follows: break ground: March, 1967; pour concrete, September, 1967; criticality, December 1970; on line, May, 1971.

Tentative plans call for the second unit to follow approximately one year later and, if convenient, Unit 1 fuel will be used in the Unit 2 startup to try to achieve an equilibrium core sooner than otherwise possible. The staff requested the applicant to look at the safety aspects of this proposal. Two safety aspects are presently suggested to my mind: (1) the startup of an untried reactor system with fuel which contains fission products and (2) the question of whether any unique core physics situations would be involved.

The applicant stated that the containment design would be based on the "stretch" capacity of 2568 MWt but that thermal analysis would be based on the initial operating value of 2452 MWt. Reactivity transients will be studied using the lower value.

The following points were brought out with respect to the site.

(1) The plant will be built 10 feet below the level of a lake to be backed up behind the planned Keowee dam. Flooding (in case of breakage of the earth dam) will have to be considered. Cooling water will be taken from the lake above a second dam and discharged to the lower lake.

(2) The Keowee dam is subject to approval by the Federal Power Commission and if permission is denied, the reactor would have to be built at a different site.

(3) Duke will have a program designed to obtain ground level meteorology at the site. (It appears that this data might not be representative until the lake is established).

(4) Emergency station power would be obtained through an underground line from a hydro plant to be built with the dam. The hydro plant containing two turbine-generators would be used for peaking loads and when not on line could be started in about one minute. There would be occasions when this power would not be available--every few years the penstock must be "dewatered" and inspected. (This could perhaps be solved by having a single diesel on site and running it continually during the outage).

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Babcock and Wilcox made a presentation which outlined the differences between the Duke plants and the reactor described in the submittal (BAW-293 and supplement) on which a preliminary review was performed by the regulatory staff. The several differences are listed in the table below.

	<u>Duke</u>	<u>B&W prelim.</u>
(1) Higher power level	822 MWe	750 MWe
(2) Operating closer to thermal limits: operating power	7% increase	-
overpower	4% increase	-
operating power DNBR (BAW-168)	1.55	1.60
(3) Higher coolant flow	131.3x10 ⁶ lb/hr	118.8x10 ⁶ lb/hr
(4) Increase in number of pins per assembly	208	200
(5) Increase in length of active fuel	144 in.	100 in.
(6) Increase in temperatures:		
inlet	550°F	540°F
outlet	603°F	591°F
(7) Steam generator (still 35°F superheat), steam pressure	925 psia	825 psia
steam temperature	570°F	558°F

Other differences from the original B&W proposal include:

- (1) The plants will share part of the engineered safeguards. Redundancy in each set of safeguards would be provided by a single "swing" unit which could be valved to either plant.
- (2) Each containment will be prestressed, post-tensioned and have an internal lume of about 2 x 10⁶ cubic feet. A non-insulated liner is planned. The containment will rest on a slab base with no rock anchors.
- (3) A common spent fuel pit will be provided.
- (4) The units will have a 25% bypass capability but will have the capability to dump a 100% load by venting secondary steam to the atmosphere.
- (5) About 90% of the safeguards will be outside the containment.

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The staff listed a number of items which should be more fully treated (than in the BSW preliminary submittal) in the hazards analysis to be prepared by Duke. These items had been previously discussed with BSW.

A. Containment

- (1) Sufficient information should be included in the application to demonstrate how the design criteria will be met. For example, containment isolation criteria should be defined and valve arrangements for each type of penetration should be supplied to show how the criteria are met.
- (2) Vessel classifications should be supplied as well as missile protection criteria.
- (3) It was stated by the staff that the design basis accident for the containment should assume the fission product release and the metal-water reaction associated with a core meltdown.

B. Reactor Systems

- (1) Complete process and instrumentation diagrams should be supplied for all systems associated with the reactor, including component cooling and service water systems.
- (2) The design bases for the primary system safety valves should be provided in terms of load rejection capability and also analyses to show that these are adequate to protect the plant.
- (3) A full description of the steam generators will be required, including design bases and experimental data which would support the once-through design.
- (4) Information on the rod drives desired by the staff includes (a) how the disengagement of the nutator gears will be insured when a scram is required, (b) how the de-energization of the magnet coils will be insured and (c) what the maximum capabilities of the drive are to move against physical restrictions in the channels.
- (5) The methods of determining (measuring) boron worth and rod worth during operation should be elaborated.

C. Thermal Design

A sensitivity analysis should be provided for the thermal design, showing the effect of variation of important parameters on the probability of fuel burnout. A comparison of the BAW-168 correlation with other current correlations should also be provided. A study such as was presented in the preliminary analysis is satisfactory.

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D. Engineered Safeguards

- (1) Separate process and instrumentation diagrams should be supplied for each engineered safeguard system including all systems required for effective operation. The systems should be detailed to the extent of a preliminary design. The injection systems provided should be designed to prevent clad melting or allow only a small amount of melting after any coolant loss accident with only emergency power available. The break sizes considered should range from small leaks to the design basis break for the containment.
- (2) All engineered safeguards systems which would transport radioactive gases or liquid outside the containment during an accident should be evaluated for potential leakage paths to the environment.
- (3) Emergency power capabilities should be defined including the minimum emergency power available during operation of one or both plants.
- (4) Minimum engineered safeguards available during operation of one or both plants should be specified, including the degree of redundancy to be available while parts of the system are undergoing maintenance.

E. Accident Analysis

- (1) The rod ejection accident should be analyzed to show that no primary system damage would result. The model used in reactivity accidents should be analyzed to determine its sensitivity to variation of all important parameters including Doppler and moderator coefficients.
- (2) Accidental dilution of the primary system from all sources of unborated water should be analyzed.
- (3) Missile damage from potential turbine failure modes should be evaluated to show that containment, engineered safeguards, and systems required for a safe shutdown would be affected.
- (4) The need for isolation valves in the secondary system to cope with postulated steam line break accidents should be evaluated.
- (5) An analysis of steam generator tube ruptures should be provided including methods for control, steps in an orderly shutdown, need for isolation valves, and resulting environment hazards.

F. Dual Reactor Installation

Systems which are to be common to both reactors should be clearly defined and the safety-related interactions of each system evaluated.

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A discussion of the reliance on cross-connections to provide redundancy in the engineered safeguards should be presented.

G. Instrumentation and Control

- (1) Test data relating to the in-core instrumentation research and development program should be provided. Discussion accompanying the data should encompass the items which B&W has indicated as included in its research and development program.
- (2) Particular attention should be given to the safety system design to ensure that no single electrical, mechanical or hydraulic failure can prevent automatic safety action, initiation of engineered safeguards or containment isolation. Live shorts at the power buses should be considered.
- (3) Sufficient instrumentation schematics should be included to allow a determination that the proposed systems conform to the Commission's proposed criteria dated November 22, 1965, numbers 15 and 16.
- (4) Consideration should be given to separating the nuclear flux servo and the (level) safety systems.
- (5) Redundancy should extend to all aspects of the power/flow safety channels including devices which measure and compare power and flow signals.
- (6) In order to evaluate the "ultimate capability" of this plant to withstand serious accidents provide the following analyses:
 - (a) Assume that all rods are simultaneously withdrawn under cold, clean conditions. The model may assume that the nuclear flux level or the high pressure trips react as designed.
 - (b) Discuss the adequacy of one boiler to remove heat following a total (external and internal) a.c. blackout.
 - (c) Discuss the manual accessibility of essential breakers under the assumption that, coincident with the design basis accident, the d.c. voltage supply is lost.
- (7) Consideration should be given to the use of stored-energy containment isolation valves actuated by fail-safe (in the event of voltage loss) instrumentation.

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- (8) Discuss the design features which assume that the control rods cannot be drawn beyond a safe maximum speed. For example, if electronic oscillators are being used, what ensures that a fault in the oscillator would not allow its output frequency (and, therefore, the rod speed) to exceed safe limits?

cc. E. G. Case
C. G. Long
D. F. Sullivan
DRL Reading
R&PRSB Reading

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