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Docket Nos. 50-424
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50-426
50-427

R. C. DeYoung, Assistant Director for Pressurized Water Reactors, L

GEORGIA POWER COMPANY - ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1, 2, 3 AND 4

Plant Name: Alvin W. Vogtle Nuclear Plant
Docket Numbers: 50-424, 425, 426, 427
Licensing Stage: Construction Permit
Responsible Branch and Project Manager: PWR-1, L. Crocker
Requested Completion Date: December 14, 1973
Description of Response: Safety Evaluation Report of the Instrumentation, Control and Electric Power Systems
Review Status: Complete

The enclosed safety evaluation report was prepared by the L:RS, Electrical, Instrumentation and Control Systems Branch. Our review was based on the Westinghouse RESAR-3, through Amendment 5, and the applicant's PSAR through Amendment 13.

Several of the applicant's designs were found unacceptable during our review. Modifications were made to include the Regulatory staff's previously established positions on RHR interlocks, accumulator valve position indication, bypass indication of inoperable status, post-accident monitoring instrumentation, cable separation criteria and diesel qualification.

Victor Stello, Jr., Assistant Director
for Reactor Safety
Directorate of Licensing

Enclosure:
Safety Evaluation Report

cc: See next page

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ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1, 2, 3, & 4

Safety Evaulation Report

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The Commission's General Design Criteria (GDC), IEEE Standards including IEEE Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE-279-1971), and applicable Regulatory Guides for Power Reactors have been utilized as the bases for evaluating the adequacy of the protection and control systems. Specific documents employed in the review are listed in the Appendix to this report.

The review of the protection and control systems was accomplished by comparing the designs with those of the McGuire Plant. Our review concentrated on those areas of design which are unique to the Vogtle plant, for which new information has been received, or which have remained as continuing areas of concern during this and prior reviews of similarly designed plants.

7.2 Reactor Trip System (RTS)

The RTS is essentially the same as that for the McGuire Plant except for that aspect of the design which permits reactor operation with one reactor coolant loop out of service. Loop isolation intelligence to the RTS is through interlocks derived from stop valves located in each loop. These interlocks in effect perform a bypass function inhibiting those protective trip functions associated with the isolated loop. We have reviewed the interlock design and concluded that insofar as it relates only to the RTS, the design meets the requirements of IEEE 279-1971 and is acceptable. Section 7.3.3 of this report discusses our concerns of loop isolation as it relates to the ESF actuation system. The RTS contains a number of anticipatory trips for which no credit is taken in the accident analysis. The applicant has been advised that the introduction of second class of trips in the RTS should not in any way result in the degradation of any primary trips. The applicant has documented in the PSAR that all anticipatory trips are designed to meet IEEE 279-1971. We have concluded that this design commitment is acceptable.

In addition to the aforementioned items, we have reviewed all other design aspects of the RTS, including functional logic diagrams, testing capabilities and control of bypasses, and concluded that this system is acceptable.

7.3 Engineered Safety Features (ESF) Initiation and Control

The ESF actuation system is functionally identical to that of the

McGuire Plant. Our review encompassed all aspects of the protection system that initiates and controls the operation of the ESF systems and their vital auxiliary supporting systems, including functional logic diagrams, testing capabilities and control of bypasses. The following sections identify those aspects of the design that were not acceptable to us and that were changed as a result of our review.

7.3.1 Manual Initiation of ESF at the System Level

Our review of the functional logic diagrams revealed that the proposed designs for manual initiation at the system level for containment spray and steamline isolation did not conform with the requirements of Section 4.17 of IEEE 279-1971.

The design of each of the two ESF actuation logic trains one manual control switch for initiating containment spray. Simultaneous operation of both switches was required to effect actuation of the sprays. The applicant has agreed, as a result of our review to modify the design to comply with IEEE 279-1971. This has been accomplished by providing two manual control switches per logic train which will satisfy both the single failure criterion and the desire to avoid inadvertent spray actuation by the operator. We considered this design change acceptable.

The design of the manual steamline isolation provided only for manual initiation at the component level. The applicant has been advised that such a design does not meet the requirements of Section 4.17 of IEEE 279-1971 which requires manual initiation of each protective action at the system level. We have requested the applicant to modify the design to conform with this standard. The applicant has committed, as a result of our review, to modify the design to comply with IEEE 279-1971, therefore, we consider this commitment acceptable.

7.3.2 Transfer to Recirculation Mode

Changeover from injection to the recirculation mode of operation following a loss-of-coolant accident was accomplished by the operator in accordance with established procedures which include a series of manual actions. The applicant has been advised that the proposed design for manual switchover to the recirculation mode does not conform to the requirements of IEEE 279-1971. In addition, the complexity of the proposed changeover procedures to be followed during worst possible operating conditions (LOCA) do not provide adequate assurance that the operator will correctly perform the required actions. We have requested the applicant to include in the design the capability for manual initiation of the ESF recirculation mode at the system level. The applicant has committed to a modified design which will provide manual initiation at the system level, therefore, we conclude that this design will meet the requirements of IEEE 279-1971 and is acceptable.

7.3.3 Reactor Coolant Loop Isolation

Reactor coolant loop stop valve position interlocks and extensive administrative controls are used to constitute the steamline break actuation circuits in a manner to recognize reactor operation with one loop out of service. Our review of the design revealed that during operation with a loop isolated, the protective logic for the active loops is effectively changed to 2/2 (high steam line differential pressure trip) which does not meet the single failure criterion. We concluded that plant operation could not be permitted with this logic arrangement. As a result of our review, the applicant has proposed Technical Specifications (requires that power to loop stop valve starters is locked out) to assure that loops are not isolated until the steam line differential pressure bistables are manually tripped. In addition, the applicant will perform an analysis to determine whether tripping of the steamline differential pressure bistables (related to the isolated loop) is required in the event of a steam line break at hot shutdown. If the analysis demonstrates that bistable tripping is required, automatic tripping of the affected bistables will be provided. We find the applicant's commitment acceptable on the basis that the design will meet all requirements of IEEE 279-1971.

7.3.4 Hydrogen Recombiner System

The hydrogen recombiner system required to function during the post-accident condition has been identified in the PSAR as an ESF system. Accordingly, the applicant, as a result of our review, has documented in the PSAR that the instrumentation, control and electrical equipment pertaining to this ESF system will be designed in accordance with the requirements of IEEE 279-1971 and IEEE 308-1971. We concluded that this design commitment is acceptable.

7.3.5 ESF Vital Supporting Systems

The applicant has identified all of the supporting systems essential to the proper functioning of the ESF. It has documented in the PSAR that the instrumentation and controls for these vital supporting systems will be designed to the same criteria as those for ESF systems that they support, including conformance with IEEE 279-1971. that this design commitment is acceptable.

7.4 Systems Required for Safe Shutdown

We have reviewed the instrumentation, control and electrical systems being provided for safe shutdown as well as the design provisions to place and keep the plant in a safe shutdown condition in the event that access to the main control room is restricted or lost. We have concluded that the designs conform to our criteria and are acceptable.

7.5 Safety Related Display Instrumentation

The design criteria for the instrumentation systems that provide information to enable the operator to perform required manual safety functions have been reviewed, and we have concluded that their design commitment is acceptable.

In addition, the applicant, as a result of our review, agreed to provide automatic bypass indication to the operator, at system level, by means of annunciator type alarm display panels. Each bypass or deliberately induced inoperable condition that affects a system that is designed to automatically perform a function important to safety shall be automatically indicated in the control room. The applicant also has stated that he intends to comply fully with the requirements of IEEE 279-1971, and Regulatory Guide 1.47. We find this commitment acceptable.

As a result of our review, the applicant has identified the instrumentation that will be available to the operator for observing the conditions in the plant and in the containment during and following postulated accidents and abnormal operational occurrences. The instrumentation will be; redundant with at least one channel recorded, qualified for the accident environment, energized from the onsite electrical power supplies, and otherwise comply with the requirements of IEEE 279-1971. We have concluded that the post-accident monitoring system proposed for the Vogtle plant is acceptable.

7.6 Other Systems Required for Safety

The applicant has documented in the PSAR the criteria that will be used in the design of the valve control circuits that will ensure the isolation of the low pressure residual heat removal system from the high pressure reactor coolant system. These criteria are consistent with those we have required in other recent construction permit reviews. The applicant also agreed to provide redundant interlocks designed to IEEE 279-1971, of diverse principles, to prevent opening and automatic closure of these valves. We have concluded that this design commitment is acceptable.

The applicant has also documented the criteria that will be used in the design for the automatic opening of the accumulator valves when either (a) the primary coolant system pressure exceeds a preselected value or (b) a safety injection signal has been initiated. We have concluded that this design commitment is acceptable.

7.7 Control Systems

The applicant has stated that the functional design of the control systems for this plant is the same as that for McGuire Plant. The applicant has not identified any differences. This commitment is acceptable and satisfies our evaluation requirements.

7.8 Periodic Testing

As a result of our review, the applicant has agreed to; provide capability for periodic testing of the response time of components affecting the time response of the reactor protection system, include the safety injection and RHR pumps in the integrated ECCS test performed during major fuel reloading shutdowns, and provide capability for periodic testing of the safety injection signal provided to trip the reactor. We have concluded that this design commitment is acceptable.

7.9 Reactor Coolant Pump Breaker Qualification

Section 15.3.4 of RESAR (complete loss of forced reactor coolant flow) assumes initiation of reactor trip and disengagement of the reactor coolant pumps from the power grid during an underfrequency condition to assure that the pumps' kinetic energy is available for full coastdown. As a result of our review, we concluded that the reactor coolant pump breakers must be qualified in accordance with the requirements of IEEE 279-1971 and be located in a seismic Category I building to assure that the pumps will disengage from their power supply when required. However, the applicant has proposed an analysis to determine the effects of underfrequency on the pumps' kinetic energy, in the event that the pump breakers failed to isolate the power supply during an underfrequency condition. The proposed analysis will consider underfrequency rates of up to 15 hz/sec. The Staff has evaluated and concluded that an underfrequency rate of 15 Hz/sec will include the most severe underfrequency transients the power grid would experience. Should the analysis demonstrate that the underfrequency conditions prevent the pumps from performing their coastdown function, the applicant has committed to qualify the reactor coolant pump breakers in accordance with the requirements of IEEE 279-1971 and, locate them in seismic Category I building. On the basis of the foregoing, we find the applicant's commitment acceptable.

7.10 Environmental and Seismic Qualification

The applicant has identified and stated that all instrumentation, control and electrical equipment important to safety will be environmentally and seismically qualified by either test and/or analysis in accordance with the appropriate IEEE Standards. We have concluded that, for electrical equipment vital to plant safety, the proposed features to be provided in the design for protection against environmental effects are acceptable.

7.11 Cable Separation and Identification for Protective and Emergency Power Systems

The applicant's criteria governing separation and installation of safety related cables were reviewed and found acceptable. As a result of our review, the applicant proposed criteria for maintaining separation of non-safety related cables from safety related buses that will include the requirements that:

(1) Non-safety related cables from one safety bus shall not be placed in the same tray with cables of a redundant safety bus.

(2) Non-safety related cables from redundant safety buses shall not be placed in the same tray.

(3) Cable armor shall not be considered as a barrier unless it is demonstrated by tests that the armor for each type of cable can serve as a substitute for physical separation.

These requirements have been documented in the PSAR. We have reviewed this area of cable separation and have concluded that the proposed criteria are acceptable.

We have also reviewed the means proposed for identification of safety related equipment, cables and cable trays, and concluded that they are acceptable.

8.0 ELECTRIC POWER

8.1 General

The Commission's GDC 17 and 18, IEEE Standards including IEEE Criteria for Class IE Electric Systems for Nuclear Power Generating Stations (IEEE 308-1971), and Regulatory Guides (RG) for Power Reactors including RG 1.6 and 1.9 served as the bases for evaluating the adequacy of the electrical power system. Specific documents used in the review are listed in the Appendix to this report.

8.2 Offsite Power System

The Offsite Power System proposed for the Vogtle Plant will consist of two (500 kV) and one (230 kV) breaker-and-a-half configuration switchyards with eight (500 kV) and three (230 kV) overhead transmission circuits connecting the switchyards to the utility grid. The transmission circuits converge on the switchyards from various directions on separate right-of-ways. Adequate separation prevents failure of one circuit causing failure of another. A combustion turbine is also connected on the (230 kV) switchyard. The high voltage circuit breakers in all switchyards are provided with primary and backup relaying circuits powered from independent supplies. The power generated in each of the four generators, units 1, 2, 3 and 4 at 22 kV, is transmitted by isolated phase bus to a main unit step-up transformer. After transformation to 500 kV, the power from units 1 and 2 will be transmitted on separate overhead transmission lines to one 500 kV switchyard and power from units 3 and 4 will be transmitted similarly to the other 500 kV switchyard.

The offsite power circuits connecting the transmission network to the onsite Class IE distribution systems for all reactor units will be provided from the (230 kV) switchyard through two physically independent circuits arranged so that failure of one will not cause failure of the other. The two circuits, through reactor unit reserve auxiliary transformers, will provide startup and shutdown power to plant auxiliaries and Class IE distribution systems. We have reviewed the proposed design and have concluded that it meets the requirements of GDC 17 and therefore is acceptable.

The continuous offsite power supply for each reactor unit non-Class IE auxiliaries is provided through two unit auxiliary transformers connected to their respective generator isolated phase bus between the generator and main unit transformer. The applicant has conducted a stability analysis on the power grid and the results showed that loss of the most critical unit on the grid will not result in loss of offsite power to the nuclear unit safety buses. We find this acceptable.

8.3 Onsite Power System

8.3.1 A-C Power System

The proposed a-c onsite power system for each reactor unit of the Vogtle Plant will have two redundant and independent 4160 volt Class IE distribution systems, with their respective 480, 240 and 120 volt load centers, which normally receive power from the (230 kV) switchyard through the unit reserve auxiliary transformers. During a loss-of-offsite power condition power to each of the redundant Class IE distribution systems will be provided by a completely independent diesel generating unit.

Each diesel generating unit will be rated for continuous operation at 6000 kW with margin in excess of the design requirements. The design basis accident load level for each of the two redundant systems will not exceed the 6000 kW continuous rating of the diesel generating unit assigned to each system. This complies with the recommendations of AEC Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies". The diesel generating units for the Vogtle Plant have not been purchased at this time. Since diesel generators of this size have not been previously qualified for use in nuclear power plants, as a result of our review, the applicant has agreed to perform qualification tests on the Vogtle units similar to those performed on the Zion 4000 kW diesel generators. The proposed qualification tests will include as a minimum the following requirements:

a. At least two tests acceptable to the staff shall be performed on each diesel to demonstrate the start and load capability of these units with some margin in excess of the design requirements.

b. Prior to initial criticality of Vogtle Unit 1, performance of at least 300 valid start and load tests. This would include all valid tests performed offsite. (A valid start and load test shall be defined as a start from design cold ambient conditions with loading to at least 50% of the continuous rating within the required time interval, and continued operation until temperature equilibrium is attained).

c. A failure rate in excess of one per hundred will require further testing as well as a review of the system design adequacy.

Independent fuel systems, complete with separate underground [illegible] three hour day tanks, are supplied for each diesel generating unit. Each underground storage tank is sized to operate required Class IE systems for a period of seven days. Each tank can serve either diesel generator.

The diesel generators are separately housed in seismic Category I structures. On the basis of our review, we have concluded that the design criteria for the a-c onsite power system meet the criteria outlined in Section 8.1 of this report, and therefore are acceptable.

8.3.2 D-C Power System

The proposed design consists of three battery systems; 125v DC system for 500 and 230 kV switchyards, 250v DC system for plant auxiliaries and the 125v DC system for safety related equipment. The 125v DC system for 500 and 230 kV switchyards is used for all units and it consists of two batteries and two battery chargers. One battery and battery charger supplies power for primary control and protective relaying, and the second battery and battery charger supplies power for back up control and protective relaying.

The 250v DC system for plant auxiliaries consists of a 250v battery and a battery charger for each unit to supply station loads which are not safety related.

The 125v DC system for safety related equipment for each unit consists of three battery chargers and two batteries. One battery and battery charger supplies power for control and instrumentation for safety related train "A" and the other battery and battery charger serves safety related train "B". The third battery charger is standby. Each charger is sized to carry its own individual load and maintain a float charge on its battery. The batteries will be separately housed in seismic Category I structures and ventilated by redundant ventilation systems. We have concluded that the design meets the criteria outlined in Section 8.1 of this report and, is acceptable.

Four redundant 120 volt vital a-c distribution buses are provided to supply power to the plant protection system instrumentation and associated circuits. Each a-c vital bus is supplied separately from an inverter. Each pair of inverters is normally supplied from separate 480 V emergency buses and upon loss of normal supply, each one of four inverters is automatically fed from its respective one of the two safety related battery units. There are also two redundant 480/208/120 volt distribution transformers, one per pair of vital buses, and each of the transformers connected to separate 480 V emergency bus. These transformers are used as an alternate source of power to the vital buses in the event of inverter

We have concluded that the d-c emergency onsite power system satisfies the requirements of GDC 17 and 18, IEEE 308-1971 and Regulatory Guides 1.6 and 1.22, and is acceptable.

APPENDIX

This Appendix lists the documents used by E. C. Marinos in the preparation of the Safety Evaluation Report for Vogtle Nuclear Stations Units 1, 2, 3, and 4.

1. 10 CFR Part 50 and Appendix A to 10 CFR Part 50.
2. Regulatory Guides 1.6, 1.9, 1.11, 1.22, 1.32, and 1.47.
3. Vogtle Units 1, 2, 3 and 4 Preliminary Safety Analysis Report (PSAR) through Amendment 13.
4. Westinghouse RESAR-3, Amendment 5.
5. The following Institute of Electrical and Electronic Engineers (IEEE) Standards:

   IEEE Std 279-1971 - "Criteria for Protection Systems for Nuclear Power Generating Stations".

   IEEE Std 308-1971 - "Class IE Electric Systems for Nuclear Power Generating Stations".

   IEEE Std 317-1971 - "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations".

   IEEE Std 323-1971 - "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations".

   IEEE Std 336-1971 -"IEEE Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations".

   IEEE Std 338-1971 - "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems".

   IEEE Std 344-1971 - "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations".

IEEE Std 379-1972 - "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems".

IEEE Std-382-1972 - "IEEE Trial Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations".

IEEE Std-387-1972 - "IEEE Trial-Use Standard: Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations".

IEEE Std 450-1972 - "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries".