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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-312 March 21, 1968

Sacramento Municipal Utilities District
Post Office Box 15830
Sacramento, California 95813

Attention: Mr. E. K. Davis
General Counsel

Gentlemen:

We have completed our review of the part of your application for a construction permit for the Rancho Seco nuclear reactor related to the reactor, engineered safety features, instrumentation, and safety analysis. The material that you have submitted does not meet our requirements for the contents of applications, as specified in 10 CFR Part 50 and elsewhere. We will not be able to continue our review of these matters until your application is complete in this regard.

Specifically, the proposed Part 50 requires coverage as fully as available information permits on the preliminary design of the facility, including the principal design criteria, the design bases and the relation of the design bases to the principal design criteria and information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

Our Guide for the Organization and Contents of Safety Analysis Reports states that information submitted should show how these principal design criteria are met by:

(1) Identifying the design bases and explaining the reasons therefor.

(2) Describing the reactor to show how the design bases have been satisfied.

(3) Showing through evaluations that design bases have been met with a reasonable margin for contingencies.

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(4) Providing a basis for such limits upon operation that might be appropriate in the interest of safety.

Other guidance is given in this document with respect to tests, inspection and surveillance.

Your information on research and development programs is incomplete; in particular the time schedule for completing the programs is not provided.

The site section does not specify the distance to the low population zone as defined in 10 CFR Part 100.

Your description of spatially dependent kinetics is incomplete. The detection system for xenon oscillations is not given in sufficient detail, nor is the proposed method for stabilizing and/or controlling potential oscillations adequately described.

The instrumentation and control section does not adequately describe the proposed systems. In particular diversification of engineered safety feature actuation signals and separation of control and safety are not adequately treated in the report.

The information we need is discussed in some detail in the attached "Request for Information." We urge that you provide full and complete answers to the enclosed request in order to minimize interruptions in the processing of your application.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Information

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See attached copies for concurrences

REQUEST FOR INFORMATION

Sacramento Municipal Utilities District
(Docket No. 50-312)

March 19, 1968

1. GENERAL

1.1 Update the discussion of your proposed design with respect to its conformance to the Commission's Proposed General Design Criteria. Include in this discussion the impact of the several design changes made in your facility.

1.2 Describe each of your research and development programs with a proposed schedule for obtaining the desired information. Include, as appropriate, when the design of the associated feature must be frozen in order to meet the schedule for construction of the Rancho Seco Plant.

1.3 If not specifically included in 1.2, describe your program, including schedule and acceptability criteria, for vibration testing of the core barrel check valves.

1.4 If not specifically included in 1.2, discuss the programs currently in progress that will assure fuel element capability for 55,000 MWD/MTU burn-up at the design power densities.

1.5 Submit the staffing and training plans for SMUD's Nuclear Project Engineering Staff.

1.6 Discuss the principal design decisions yet to be made that require nuclear and steam plant knowledge and which affect nuclear power plant safety. Indicate the approximate dates by which these decisions must be made and to what extent reliance will be placed upon contractors for making decisions. Indicate how the training plans for SMUD personnel are orientated toward these requirements.

1.7 Your Amendment No. 1 provided the SMUD response to applicable questions raised during the review of a similar plant (Metropolitan Edison). This response used information that was available through November, 1967. Please update your response to these questions by considering applicable information that became available in January, 1968.

2. SITE AND ENVIRONMENT

2.1 An analysis should be presented which relates primary coolant activity, assumed leakage rate from the primary to secondary system, removal and cleanup mechanisms for the secondary coolant, and the derived activity contained in the secondary system.

2.2 The PSAR description of the steam generator tube accident includes an assumption that the iodine water to air partition factor is 10,000. Show how this factor was derived, and indicate how concentration, temperature, pressure, and air to water volume ratio which exist throughout the course of the accident may effect this partition factor.

2.3 The PSAR calculations of off-site doses due to release of noble gases include an assumption that the average effective energy per disintegration of noble gases is 0.4 MEV. The origin or justification of that assumption should be provided.

2.4 Submit a listing of the radioactive isotopes and maximum activities of each which may be present in the liquid waste holdup tanks at any one time, and include an analysis demonstrating that failures in the liquid waste system would not cause excessive release of radioactive liquids to the environs.

2.5 Specify the distance to the low population zone as it is defined in 10 CFR Part 100, Section 100.3 (b).

2.6 Based on data presented in the PSAR, it appears that the Rancho Seco site is subject to a high frequency of inversion conditions with low transport winds. Data presented show a computed frequency of about 25% extremely stable conditions with an average wind speed of 0.9 meters per second; it would appear appropriately conservative to use this condition for calculating the 2 hour off-site doses. Please provide the environmental consequences of hypothetical accidents using this basis.

2.7 Discuss the water flow patterns in the vicinity of the plant and their associated consequences on plant operations following a failure of the on-site water storage facilities.

2.8 Provide a map of earthquake epicenters within a radius of 200 miles showing all earthquakes of intensity V or greater at the epicenter.

3. <u>REACTOR</u>

3.1 Discuss your plans for providing a negative moderator coefficient of reactivity throughout core-life in the event detailed studies show this to be a design requirement.

3.2 Describe your derivation of the "power doppler coefficient" given in Table 3.2-3 of the PSAR and compare the time constant of this coefficient with that of the system in the analytical model.

3.3 Submit the latest available results of those analyses on xenon oscillations described on pages 3.2-21 and 3.2-23 of the PSAR and specify the dates when the remaining analyses will be completed.

3.4 Discuss the detection system for xenon oscillations and indicate the expected minimum sensitivity of this system during power operation.

3.5 Describe the 2-dimensional analysis method for evaluation of xenon instabilities.

3.6 Assuming that control rods are used to stabilize xenon oscillations, give the maximum values anticipated for the transient and steady-state errors in local power density at the hot spots.

3.7 Indicate the margin of xenon stability by giving the power level at which xenon oscillations are predicted to occur at various times during core life.

3.8 Discuss the fuel management plans and techniques that will limit maximum fuel burn-up to 55,000 MWD/MTU and describe the associated uncertainties.

3.9 Discuss your calculational model and indicate the error band on the fast neutron flux ($E_n > 1.0$ Mev) at the pressure vessel inner surface which was calculated to be 3.4×10^{10} (n/cm^2 - sec). Include in the discussion:

a) How azimuthal variations are treated in the analysis and relate these to the azimuthal placement of the surveillance specimens.
b) The uncertainties associated with the attenuation factor of $6.0 \times 10^{13}/3.4 \times 10^{10}$ or 1760 and relate their potential consequences to higher values of NDTT for the pressure vessel wall.
c) The maximum fast neutron exposure (see pg. 4.1-8) is indicated to be 3.0×10^{19}(n/cm^2) or, at 80% load factor, 1.9×10^{10} (n/cm^2 - sec). Explain the relationship between this design limit and the data given in Table 3.3-7 of the PSAR with respect to the factor of 2 conservatism indicated on page 3.2-14 of the PSAR.

3.10 Discuss the probability for a single fuel pin to undergo DNB during the first three years of power operation at rated conditions. (Alternatively, specify the number of fuel pins that have greater than 50% probability for undergoing DNB during three years of power operation at rated conditions). Include in your discussion:

a) The potential consequences of a single fuel pin undergoing DNB during full power operation.
b) The time behavior of events that occur in those fuel pellets located in the vicinity of the DNB surfaces.

c) Definition of the word "jeopardy" as used in the PSAR to describe the conclusions of your statistical analyses.

4. REACTOR COOLANT SYSTEM AND OTHER CLASS I SYSTEMS

4.1 Thermal Shock

With regard to thermal shock on reactor components, induced by operation of the emergency core cooling system (ECCS), provide details of an analysis which indicates that the reactor vessel and reactor internals can withstand the rapid temperature change at the end of their design life. The analysis should include both the ductile yielding and the brittle fracture modes of failure.

4.1.1 The brittle fracture analysis for the vessel should assume an initial crack size just below the critical crack size corresponding to the stresses present during normal operation and transients. Since the initial crack is most likely to exist in a weld or a heat affected zone, the analysis should consider two cases: a circumferential crack, and a crack parallel to the axis of the reactor vessel. The details of the analysis should be provided including specific information on:

(a) The critical stress intensity factor (K_{IC}) assumed, and the basis for its selection,
(b) The assumed time-integrated neutron flux (nvt) at the reactor vessel inner diameter,
(c) The value of residual stresses assumed in the base metal and the weld areas,
(d) The initial crack geometry and size assumed in the analysis,
(e) Equations used to correlate crack size with the calculated stress intensity factor (K_I).

4.1.2 The details of the ductile yielding mode of analysis for the vessel should include the following information:

(a) The geometry of the plate and the cooling method assumed in the analysis,
(b) The heat transfer coefficient used, its experimental basis, and the degree of conservatism involved,
(c) The initial temperature of the vessel as a function of time delay in injecting the cold water,
(d) The effect of axial temperature gradient in the vessel, during filling with cold water, on the total stress intensity and the distortion of the vessel,
(e) The temperature profiles and the calculated thermal stress profiles through the thickness of the plate for several times during the cold water injection transient,

(f) The magnitude of the axial dead load stresses in the vessel,
(g) The magnitude of the stresses in the vessel shell due to potential simultaneous seismic loading,
(h) The value of the yield stress used as the failure criterion in the ductile yielding analysis.

4.1.3 Based on the analyses for the vessel provide:

(a) An estimate of the maximum acceptable initial temperature of the vessel that could be tolerated without failure of the vessel,
(b) An estimate of the maximum neutron flux exposure (nvt) of the vessel that could be tolerated without vessel failure,
(c) An estimate of the maximum allowable pressure stress, when combined with other stresses present in the vessel, which could be tolerated without failure.

4.1.4 Evaluate the capability of the piping, safety injection nozzles, and vessel nozzles to withstand the transient.

4.1.5 Evaluate the effects of this transient on the core barrel and other internals with regard to assuring that distortion would not restrict the flow path of the emergency core coolant.

4.2 Seismic Design

4.2.1 For all Class I systems and components provide the design basis load combinations and the proposed stress and deformation limits for each combination.

4.2.2 Supply criteria or specific information on the interaction forces, deformation and stresses connected with the relative motions between the reactor vessel, steam generators or other large components. Indicate how these relative motions will be controlled by snubbers or other means, and what reaction forces (and corresponding stresses) will be transmitted to the pipes.

4.2.3 Identify specific reactor internals which must maintain their functional performance capabilities to assure safe shutdown of the reactor. Provide calculated (or estimated) maximum limits of deformation or stress, at which inability to function occurs, for each component identified. Also, supply the calculated (or estimated) maximum design limit value, and the expected deformation or stress. In all cases identify the applicable loading combination and state the proposed margin of safety.

4.2.4 For reactor internals provide information that will permit evaluation of the effect of irradiation on the material properties and on the proposed deformation limits.

4.3 Discuss the full power radiation environment with respect to corresponding damage thresholds for the control rod actuators and the primary loop pumps and pump motors. Consider the N-16 activity, the fission product activity in coolant, and the radiation streaming contributions.

4.4 Provide a tabulation of all the nuclear pressure vessels in the Class I (seismic design) systems in the facility. The tabulation should include a notation of whether the vessel design is complete, the stage of fabrication of the vessel, and the extent to which each of the vessels will comply with each of the 34 supplementary criteria in "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels", issued by AEC Press Release No. IN-817, dated August 25, 1967.

For each vessel, provide a discussion that represents the reason why total compliance is not feasible for each criterion not met in its entirety.

4.5 Submit Certified Code Design Specifications for component parts of the Class I systems as required by the ASME Code Section III, paragraph N-141 (passed 6-23-67).

6. ENGINEERED SAFETY FEATURES

6.1 Update your PSAR with those design revisions described at our March 1, 1968 meeting.

6.2 Provide the anticipated post-accident radiation dose levels in the containment. Compare the anticipated gamma exposures with damage thresholds for the engineered safety features.

6.3 Describe the test programs that will assure adequate performance of the engineered safety features in the post-accident environment.

6.4 Provide an evaluation of the ultimate iodine removal capability for the proposed spray systems that can be rigorously supported by presently available experimental evidence. Include a discussion of spray effectiveness in removing aerosols.

6.5 Provide an analysis of the physical aspects of the proposed spray systems, including the fraction of the entire containment volume directly covered by the sprays, the convection circulation into the spray pattern, the range of drop sizes, and the relative temperatures of spray and containment air with their effect on iodine removal rate and efficiency.

6.6 Discuss the extent to which reversible, competitive, and slow chemical reactions have been considered in the evaluation of the effectiveness of the spray systems. Consider the contribution of liquid film mass transfer resistance in the calculation of the overall mass transfer coefficient.

6.7 Provide an analysis of the composition and pH of the emergency core cooling solution as a function of time following the design basis loss-of-coolant accident. Consider spray system additives, soluble neutron poisons, fission and corrosion products, elements leached from concrete, etc.

6.8 Provide a discussion of the extent to which exposure to the solution discussed in item 6.7 above will be factored into the procedure for selection of materials for the engineered safety features for the facility. Discuss the systems that will be affected and the nature of the considerations that will be taken into account.

6.9 Discuss the time, temperature, and radiation dependent stability of the spray solution under both storage and post-accident recirculating conditions and indicate the possibility of forming solid decomposition products or precipitates which could potentially interfere with system performance.

6.10 Discuss both the time-dependent radiolytic and chemical hydrogen formation under post-accident conditions for the solution given in item 6.7 above. Include an estimate of total γ and β activity in both the core and in the liquid, and of the total expected irradiation dose characteristics. Indicate the extent of hydrogen formation by chemical reaction (corrosion) with exposed reactor materials.

7. INSTRUMENTATION AND CONTROL

7.1 Discuss and evaluate the differences between the SMUD Station, Babcock & Wilcox designed protection systems which initiate reactor trip and engineered safety feature action and those to be incorporated in the Three Mile Island Station (Docket No. 50-289). The discussion should include preliminary design of the complete circuit from sensors to actuation logic.

7.2 With respect to the reactor protection and engineered safety feature actuation circuits to be designed by other than Babcock and Wilcox, identify the design features which differ from the proposed IEEE standard for Nuclear Power Plant Protection Systems. Justification for all differences should be provided.

7.3 Describe and evaluate the criterion to be used in providing for the physical identification of the reactor protection and engineered safety feature equipment including panels, components, and cables.

7.4 Describe and evaluate the changes which will be made in the design of the instrumentation and control systems as a result of the ACRS recommendations contained in the Three Mile Island letter. Include in the discussion:

(a) Diversity of engineered safety feature actuation signals and

(b) Separation of control and protection systems.

7.5 Identify the instrumentation and electrical equipment which must function in an accident environment. Discuss and evaluate the qualification testing which is necessary to insure that this equipment will function in the accident environment. Your intentions with respect to obtaining the required data should be discussed.

7.6 With respect to the reactor protection and engineered safety feature signals which feed annunciators and/or a data logging computer, describe and evaluate the design criterion to be used to assure circuit isolation.

7.7 Identify and discuss the differences between the SMUD Station, Babcock and Wilcox designed control systems and those to be incorporated in the Three Mile Island Station (Docket No. 50-289). This discussion should include an evaluation of the safety significance of each system.

7.8 Identify, discuss, and evaluate the differences between the SMUD Station in-core instrumentation and that to be incorporated in the Three Mile Island Station (Docket No. 50-289).

7.9 Describe the control room ventilation system and evaluate the need for placing the system automatically in a recirculation mode utilizing an airborne radiation detector which monitors the intake duct.

8. <u>ELECTRICAL SYSTEMS</u>

8.1 In the evaluation of the ability to supply power to engineered safety features from offsite sources, consider the effect of the sudden tripping of the unit. In addition to the effect on system stability, consider coincident failures in the generating station switchyard to assure that none will cause the loss of all offsite power to the station. Consideration should be given to but not be limited to the following: faults, circuit breaker failures, control circuit failures, and battery failures.

8.2 Evaluate the ability of the offsite power to meet General Design Criterion 39 with the proposed single startup transformer.

8.3 Describe and evaluate the automatic loading sequence for the emergency diesel generators.

8.4 Provide an evaluation of loads (HP) required to be powered in the interest of safety and the relationship of the maximum emergency load that may be placed on each diesel generator to the rating (KW) of the generator.

8.5 Describe and evaluate the provisions to prevent two diesel generators from being connected together and from being connected to another source of power that is out of phase.

9. AUXILIARY AND EMERGENCY SYSTEMS

9.1 Submit the design revisions for the cooling water systems that were described at our meeting on March 1, 1968.

9.2 Discuss the maximum extent (frequency and duration) to which reservoir make-up water will be used in the event of canal water supply system outage.

9.3 Discuss the plant's capability for detecting fuel failure. This discussion should include the detection time as a function of fuel failure severity.

9.4 Submit a brief statement of your provisions in the emergency cooling water supply to cope with the lowest anticipated ambient temperatures ($\sim 19^{\circ}$ F).

9.5 Discuss the provisions for draining the spent fuel pool.

9.6 Discuss the potential for inadvertant draining of the spent fuel pool.

9.7 Discuss the potential for draining the water in the fuel transfer canal and tube and specify the required fission product decay period after which the fuel elements do not require water cooling.

12. CONDUCT OF OPERATIONS

12.1 Discuss further the relationships between SMUD, Bechtel, B&W, WEC, and others. This discussion should include a list of the subsystems and support functions provided by the principal parties.

12.2 Provide organization charts that show the contributions by SMUD, Bechtel, B&W, WEC, and others during the construction phase and the operations phase.

12.3 Expand organizational charts in the PSAR to show lines of responsibility for quality control efforts during the construction phase.

12.4 Submit an organizational chart for the Bechtel Corporation indicating responsibility channels for Quality Assurance and Quality Control efforts for this project. Delineate home office as well as site groups.

12.5 Submit an organizational chart for the Babcock and Wilcox Company indicating responsibility channels for Quality Assurance and Quality Control efforts.

12.6 Submit the staffing and training plans discussed at DRL on February 5, 1968 for the Rancho Seco No. 1 operating personnel.

13. INITIAL TESTS AND OPERATIONS

13.1 Discuss the extent to which test results will be documented.

13.2 Discuss your plans for measuring and/or verifying the threshold conditions for xenon oscillations. Include in your discussion the extent to which data from earlier plants will be used.

13.3 Provide a det 'led outline of the test program for each engineered safety system. The ou..ine should provide a set of test objectives for each system, a brief description of the proposed test, and a brief discussion on how achievement of design objectives can be ass ired.

13.4 Provide the following information in outline form regarding emergency planning for the SMUD facility:

(a) Plan objective,
(b) Scope,
(c) Delineation of responsibility and authority for plan implementation,
(d) Notification liaison to be established with federal, state and local authorities and emergency assistance personnel that they provide.
(e) Provisions made with local hospital and physicians for treatment of injured persons, including contaminated persons.
(f) Instrumentation to be installed with readouts in the control room to be used for assessment of the extent of a radioactive release, both on site and offsite.
(g) Proposed training of onsite staff and means to be used to evaluate the plan's effectiveness on a periodic basis.

14. SAFETY ANALYSIS

14.1 Describe the analytical model used to study the reactor system response to a 100% loss of demand load and to total loss of a.c. power.

14.2 Provide the following results of your analysis of the load loss transient:

a) Rise in average moderator temperature,
b) Minimum DNB ratio during the transient,
c) Rise in reactor loop pressures,
d) Extent of turbine over-spe ,
e) The reactor thermal power transient, and
f) Fuel and clad temperatures.

14.3 Describe the natural circulation characteristics of the primary loop system. Will operation of primary loop relief valves, due to its dead-band characteristics, affect this flow?

14.4 In Figures 14.2.1 through 14.2.11 of the PSAR, the reactor kinetic parameters are given for α_d, α_M, ℓ^*, and τ. What were the corresponding values for βeff?

14.5 Discuss the technique used in calculating the effective delayed neutron fraction and include a summary of your calculations for the end-of-life value.

14.6 Discuss the accuracy of the energy yield predictions for the rod ejection accident. Your discussion should include the anticipated power profile transients.

14.7 How is spatial dependence treated for β eff, k_{∞}, 1*? Evaluate the uncertainty in peak power densities associated with this approach.

14.8 For the rod ejection accident(Section 14.2.2.2 of the PSAR), discuss the predicted pressure pulse in the reactor vessel and the associated uncertainties.

14.9 Discuss the potential for reactivity insertion and the associated consequences when a repaired pump is returned to service.

15. TECHNICAL SPECIFICATIONS

15.1 Identify those items that will eventually be classified as technical specifications that now affect plant design. Examples include the minimum conditions of operation on: engineered safety features; emergency generators; and in-core flux monitors.

16. RESPONSE TO ACRS LETTER ON METROPOLITAN EDISON'S PLANT

16.1 Describe and evaluate the design changes that will be made in the reactor scram system as a result of the ACRS recommendation contained in the Three Mile Island letter regarding potential failure to de-energize the scram bus.

16.2 Discuss and evaluate your design changes that will provide the capability for prompt detection of gross failure of a fuel element.

16.3 Discuss and evaluate your program of analysis and design directed to assure that fuel failures will not significantly inhibit the ECCS from preventing clad melting.

16.4 Discuss your analysis and design efforts for use of part length rods to control potential axial and diametral xenon oscillations. Include in this discussion a description of your latest design concept and its estimated performance characteristics.

16.5 Discuss the effect of blowdown forces on reactor internals by identifying appropriate load combinations and deformation limits.

16.6 Discuss and evaluate your program to experimentally study vibrations in the check valves.