SAXTON NUCLEAR EXPERIMENTAL CORPORATION

DOCKET NO. 50-146

LICENSE DPR-4

CHANGE REQUEST NO. 26

1. Applicant hereby submits Change Request No. 26 in compliance with paragraph 3.B of license DPR-4 for change of the Technical Specifications to be authorized by the Commission as provided in 10 CFR 50.59.

SAXTON NUCLEAR EXPERIMENTAL CORPORATION

By /s/ R. E. Neidig
President

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3. Safety Considerations

The portion of the loop under consideration is approved by the Pennsylvania Department of Industrial Safety under Section I of the ASME Code. A Pennsylvania Special Number, No. 2162, is assigned to the loop since, to comply with nuclear requirements, certain components were fabricated under the rules of Section VIII of the ASME Code and the applicable nuclear code case, 1273N.

The code approved safety valves for the loop are PSV-1 and PSV-2. These valves are located downstream of the pressure tube, and are now set to actuate at nominal pressures of 4000 psig and 4120 psig, respectively. These set pressures are based on a pressure tube design pressure of 4000 psig. The location and setting of these two valves fulfill the requirements of the Code with respect to overpressure protection for the pressure tube and the loop. Relief valve, PSV-3, which is not required by the Code is installed to provide additional overpressure protection for the loop.

Lowering the settings of PSV-1 and PSV-2 to a maximum value of 3980 psig takes into consideration the pressure drop which occurs across the pressure tube during normal loop operation. This pressure drop is calculated to be a maximum of 140 psi. The new settings of 3980 psig will limit the pressure at the tube inlet to a maximum of 103% of the tube design pressure as permitted by Section VIII of the ASME Code.

Lowering the setting of PSV-3 to 4800 psig meets all the requirements of Section I of the ASME Code. In addition, this change still allows adequate operating flexibility between the safety valve setting and loop operating pressure, and is consistent with lowering the settings of PSV-1 and PSV-2.

4. Health and Safety

It is our conclusion that the health and safety of the public will not be endangered by this change.

Saxton

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USAEC HQS GTWN
WU WSH
TLXA036 TLX040 (23)AA118 PD256
P WAA036 NL PD TDACX SAXTON PENN APR 11
D J SKOVHOLT, ASST DIRECTOR FOR REACTOR OPERATIONS-DIVN OF REACTOR
LICENSING UNITED STATES ATOMIC ENERGY COMMISSION
For Div of Compliance
WASHDC
REFERENCE: DOCKET NO 50-146, DPR-4, TECHNICAL SPECIFICATIONS,
CHANGE NO QYN SUPERCRITICAL TECHNOLOGY PROGRAM. AND APPROVAL
REQUESTED TO CHANGE TECHNICAL SPECIFICATIONS AS FOLLOWS: CHANGE
NO QY SECTION C.3, ZPRESSURE RELIEF" CHANGE NOMIAL TO MAXIMUM

C.3.A: CHANGE PSV-1 SETTING FROM 4200 TO 3900.
C.3.A; CHANGE PSV-2 SETTING FROM 4120 TO 3900
C.3.B; CHANGE PSV-3 SETTING FROM 5000 TO 4800 PURPOSE:
REDUCE SET PRESSURES OF SAFETY RELIEF VALVES TO CONFORM WITH
APPLICABLE ASME CODES.
SAFETY CONSIDERATIONS: THE CHANGE TO PSV-1 AND PSV-2 SETTINGS

TAKS INTO ACCOUNT MAXIMUM CALCULATED FRICTION PRESSURE DROPS
SO THAT ASME CODE ALLOWABLE PRESSURES ON THE LOWEST RATED COMPONENT
WILL NOT BE EXCEEDED. PSV-3 SETTING IS LOWERED TO 4800 SO THAT
ASME CODE REQUIREMENTS CAN BE MET EVEN FOR ABNORMAL SITUATIONS
AND IS CONSISTENT WITH LOWERING THE SETTINGS OF PSV-1 AND PSV-2.

IT IS OUR CONCLUSION THAT THE HEALTH AND SAFETY OF THE
PUBLIC WILL NOT BE IN DANGER BY THIS CHANGE.
APPROVAL IS REQUESTED BY APRIL 17 QOYU TO AVOID DELAY
OF EXPERIMENTAL TESTS PROGRAM
SAXTON NUCLEAR EXPERIMENTAL CORP, R E NEIDIG PRES

50-146 DPR-4 NO 16 AEC NO 16 C.3 C.3.A PSV-1 RWPP EOIP C.3.A
PSV-2 RQWP EOIP C.3.B PSV-3 TPPP RIPP PSV-1 PSV-2 PSV-3 RIPP

PSV-1 PSV-2 QU QOYUM
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C. Variable Frequency Motor Generator Set Modifications

1. What prevents the reactor coolant pump from operating from the 440 volt bus at high power? Could the underfrequency reactor trip be made ineffective by operating the MG set at 63 cycles with the generator breaker (52/VFG) open?

2. What are the consequences of a loss-of-flow accident from maximum reactor power caused by inadvertent opening of the MG set generator circuit breaker 52/VFG?

D. Emergency Core Cooling Controls

1. Could a single failure in the flow totalizer keep the pumps from starting or stop them prematurely?

2. Could the injection block switch disable all safety injection? Can a single operator error or a single switch failure disable safety injection?

III. Thermal-Hydraulic Design

Your presentation of the thermal and hydraulic capability of the core design consists principally of evaluations of steady state and transient DNB ratios and fuel temperatures for the hottest core location. A complete assessment of the conservatism or safety of the design requires some understanding of the condition of the entire core during normal and transient situations so that we can evaluate the margins available before large numbers of fuel rods exceed design limitations. Thus, our evaluation of the design must be based on the overall core condition, as well as that of the so called hot spot. A presentation using these considerations should be made as follows:

A. Prepare a distribution curve showing the fraction of the core (and number of rods) operating at the various power levels for design and overpower conditions.

B. Using the appropriate DNB correlation and the above distribution, determine the corresponding DNB ratios and the statistical number of fuel rods that could experience DNB.

C. Perform an uncertainty analysis by arbitrarily assuming certain errors in major parameters used in calculating the number of rods experiencing DNB. For example, calculate the number of rods with DNB, as a function of possible percentage errors in the DNB correlation, power distributions, flow rates, and power levels.

IV. Fuel and Materials

A. In view of the anticipated high burnups and power levels, is it reasonable to assume that fuel swelling will not occur at a higher rate than 0.0016 $\Delta V/V$ per 10^{20} fs/cc as shown in Figure 7 of Reference 1?

B. What will be the temperature distribution in a fuel element in a high flux position? What will be the temperature gradient across the cladding, across the gap and in the fuel itself if the density of the fuel is varied between some specified limits?

C. What are the data that support the statement that tube-reduced Zircaloy would not fail for 7.4 years at the maximum of 16000 psi expected design stress.

D. What are the applicable composition, dimensions and inspection specifications for zirconium alloy tubing and the end closure that make sure that the expected design stresses cannot be exceeded?

E. Under normal conditions stresses in the cladding will be compressive and strains will be limited by the fuel pellet; but what will be the stress and strain pattern in a sudden depressurization followed by a scram when fuel temperatures are high and fission gas pressure is effective?

F. Is there experimental data on hydrogen absorption by the cladding that go beyond the burnup and power level of Reference 3? What would be the effect of hydrogen absorption by the cladding on its fatigue properties, and what would be the combined effect of hydrogen absorption and irradiation embrittlement?

V. Test Assemblies

Provide (1) comparison of the operating conditions for the test assemblies versus previously approved conditions at 23.5 Mw and (2) safety evaluations for any new steady state or transient conditions.

VI. Rod Ejection Protection

Provide a description and drawing of the mechanical st ... on the control rods which are designed to prevent ejection accident. Include an analysis to demonstrate that these stops would not fail in the event of a complete shearing of a control rod nozzle.

VII. Dose Calculations

By use of the methods and parameters of TID-14844, we find that an exclusion radius of at least 300 meters is needed for 35 MW operation. Your calculations indicate that the exclusion radius should be at least 270 meters. Compare the assumptions used in your calculations to those used in TID-14844. Which factors contribute most to your lower dose estimates?

VIII. Power Escalation Program

Provide additional detail on your program for power escalation. Include in your discussion, the size of each step in power increase (and specific power increase), the length of time at each power and the measurements and evaluations required after each step.

IX. Emergency Core Cooling

In our letter to you dated December 30, 1966, we requested that you consider the emergency core cooling provisions to determine the need for additional provisions to limit the release of fission products from the core. In addition, we requested that you perform suitable analyses to determine the degree to which emergency core cooling must be relied upon to maintain containment integrity. Please inform us of the progress made thus far and your schedule for submittal of the reply.