SNUPPS
Standardized Nuclear Unit
Power Plant System

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Nicholas A. Petrick
Executive Director

August 14, 1981
SLNRC 81-69 File:0541
SUBJ: NRC Request for Additional Information - Materials Engineering

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos. STN 50-482, STN 50-483, and STN 50-486

Reference: 1. NRC (Tedesco) letter to UE (Bryan), dated July 28, 1981, same subject
2. NRC (Tedesco) letter to KGE (Koester), dated July 28, 1981, same subject

Dear Mr. Denton:

The referenced letter requested additional information for the SNUPPS FSAR. Enclosure 1 to this letter provides either the requested information or a schedule for providing the information. Concerning items #123.8 and #123.9, WCAP-9842 is being forwarded to the NRC by a separate letter. Concerning item #123.11, Enclosure 2 is a copy of the UT examination procedure. The PT examination procedure for the reactor coolant pump flywheels is under development. Enclosure 1 will be incorporated in a future FSAR Revision.

Very truly yours,

R L Stright for

Nicholas A. Petrick

Enclosures
cc: J.K. Bryan UE
D.F. Schnell UE
G.L. Koester KGE
D.T. McPhee KCPL
W.A. Hansen NRC/Cal
T.E. Vandel NRC/WC

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Q123.1 Identify whether SA-540 Class 1 or 2 material was used for closure bolting in the reactor coolant pumps. If SA-540 Class 1 or 2 materials were used for closure bolting in reactor coolant pumps, demonstrate the generic adequacy of the fracture toughness and demonstrate compliance with Paragraph I.C of Appendix G, to 10 CFR Part 50.

RESPONSE

SA-540 Class 1 or 2 material was not used for closure bolting in the reactor coolant pumps for the SNUPPS units.

Q123.2 Indicate whether the individuals performing the fracture toughness tests were qualified by training and experience and whether their competency was demonstrated in accordance with a written procedure. If the above information cannot be provided, state why the information cannot be provided and identify why the method used for qualifying individuals is equivalent to those of Paragraph III.B.4 Appendix G, 10 CFR Part 50.

RESPONSE

The response to this question is planned for submittal by August 31, 1981.

Q123.3 To demonstrate compliance with the beltline material test requirements of Paragraph III.C.2 of Appendix G, 10 CFR Part 50:

a. Provide a schematic for the reactor vessel showing all welds, plates and/or forgings in the beltline. Welds should be identified by shop control number, weld procedure qualification number, the heat of filler metal, and type and batch of flux. Provide the chemical composition for these welds (particularly Cu, P, and S content).

b. Indicate the post-weld heat treatment used in the fabrication of the test welds.

c. Indicate the plates used to fabricate the test welds.

d. Indicate whether the test specimen for the longitudinal seams were removed from excess material and welds in the vessel shell course following completion of the longitudinal weld joint.

RESPONSE

The response to this question is planned for submittal by August 31, 1981.

Q123.4 To demonstrate compliance with the fracture toughness requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50:

a. Provide the RT_{NDT} for all RCPB welds which may be limiting for operation of the reactor vessel.

b. Indicate whether there are any RCPB heat-affected zones which require CVN impact testing per paragraph NB-4335.2 of the 1977 ASME Code. Provide CVN impact test data for these heat-affected zones which ma, - limiting for operation of the reactor vessel.

c. Indicate that there are no ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code. If there are ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code, provide CVN impact and drop weight data for all materials which will be limiting for operation of the reator vessel.

RESPONSE

The response to this question is planned for submittal by August 31, 1981.

Q123.5 Revise the FSAR to indicate that the conclusions of Westinghouse Topical Report WCAP 9292 are applicable to Callaway Unit 1 (Wolf Creek) SA-533 Grade A, Class 2 steel and SA 508 Class 2a steels.

RESPONSE

The conclusions of Westinghouse Topical Report WCAP-9292 are applicable to the SNUPPS units. Refer to Section 5.2.3.3.1.

Q123.6 Provide actual pressure-temperature limits for Callaway Unit 1 (Wolf Creek) based upon the limiting fracture toughness of the reactor vessel material and the predicted shift in the adjusted reference temperature, RT_{NDT}, resulting from radiation damage. The pressure-temperature limits for the following conditions must be included in the technical specifications when they are submitted:

a. Preservice hydrostatic tests,
b. Inservice leak and hydrostatic tests,
c. Heatup and cooldown operations, and
d. Core operation.

RESPONSE

The pressure-temperature limits will be forwarded by a separate letter by August 24, 1981. Proposed Technical Specifications are under preparation and are planned for submittal in October 1981.

Q123.7 Provide full CVN impact curves for each weld and plate in the beltline region. Provide the data in tabulated and graphical form.

RESPONSE

The response to this question is planned for submittal by August 31, 1981.

Q123.8 To demonstrate the surveillance capsule program complies with Paragraph II.C.3 of Appendix H:

a. Provide the withdrawal schedule for each capsule

b. Provide the lead factors for each capsule.

c. Indicate the estimated reactor vessel end of life fluence at the 1/4 wall thickness as measured from the ID.

RESPONSE

The requested material is provided in WCAP-9842 for Callaway Unit 1. Similar reports for Wolf Creek Unit 1 and Callaway Unit 2 will be available later.

Q123.9 Identify the location of each material surveillance capsule and the materials in each capsule.

a. For each base metal and heat-affected zone surveillance specimen provide the specimen type, the orientation of the specimen relative to the principal rolling direction of the plate, the heat number, the component code number from which the sample was removed, the chemical composition especially the copper (Cu) and phosphorus (P) contents, the melting practice and the heat treatment received by the sample material.

b. For each weld metal surveillance specimen provide the weld identification from which the sample was removed, the weld wire type and heat identification, flux type and lot identification, weld process and heat treatment used for fabrication of the weld sample.

c. Provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

RESPONSE

The requested material is provided in WCAP-9842 for Callaway Unit 1. Similar reports for Wolf Creek Unit 1 and Callaway Unit 2 will be available later.

Q123.10 Indicate the normal operating temperature of the flywheels and provide CVN impact and drop weight test data from each flywheel that indicates the RT_{NDT} of the flywheels are 100°F less than their normal operating temperatures.

RESPONSE

The response to this question is planned for submittal by August 31, 1981.

Q123.11 Submit for review an inservice inspection program for the pump flywheels which complies with Paragraph C.4 of Safety Guide 14, October 27. 1971.

RESPONSE

The inservice inspection program for reactor coolant pump flywheels for Callaway and Wolf Creek is in compliance with the ASME Code and with the recommendations of Regulatory Guide 1.14, Revision 1, August 1975. This Regulatory Guide is addressed in Appendix 3A of the FSAR and no exceptions are taken to paragragh C.4.