DEC 0 5 1990

Project No. M-53

Pacific Sierra Nuclear Associates
ATTN: Dr. John V. Massey
General Manager
5619 Scotts Valley Drive
Scotts Valley, CA 95066

Dear Dr. Massey:

We have received comments from the Nuclear Regulatory Commission's Performance and Quality Evauation Branch (PQEB) that your responses dated October 31, 1990, addressing the Pacific Sierra Nuclear Associates (PSN) Quality Assurance (QA) process in the design control of the Ventilated Storage Cask System (VSC) appear to be adequate and acceptable. Therefore, no further response is required.

While we are in the midst of preparing the Safety Evaluation Report (SER), there are still some technical issues that need to be satisfactorily resolved before we can complete our review and issue an SER. See enclosed comments. Furthermore, please provide us Revision 3 of "Topical Report (TR) for the VSC" which should incorporate the latest changes, corrections, additions, deletions, etc., in all the PSN TR related documents and drawings to enable us to perform a thorough final review of the PSN TR.

If you have any questions, please call K. C. Leu 301-492-0696.

Sincerely,

Original Signed by
John P. Roberts

John P. Roberts, Section Leader
Irradiated Fuel Section
Fuel Cycle Safety Branch
Division of Industrial and
Medical Nuclear Safety

Enclosure:
Comments

bcc: JStokley, SAIC
DISTRIBUTION:

Project M-53	PDR	NRC File Center	IMNS Central File
NMSS R/F	IMIF R/F	IMSB R/F	FLBrown
JFRoberts	KCLeu	FSturz	JSpraul
CJHaughney	DKasun	SGramman	

[KCL/PSNA]

OFC	:IMIF	:IMIF	:IMIF
NAME	:KCLeu:fb	:FBrown	:JRoberts
DATE	:11/30/90	:12/5/90	:12/5/90

OFFICIAL RECORD COPY

NF09

I. Design Comments

1. PSN examined the outer steel shell of the MTC in the vicinity of the trunnion. They provided an argument to show that the relative stiffness of the inner and outer shell of the MTC was very much larger than the stiffness of each shell out of plane. The purpose was to justify not modeling any local bending of the shell. However, the NRC staff does not concur with the PSN argument or conclusion. The NRC staff estimates that the two shells acting as flanges have a relative stiffness of only 1.47 times the weld area of both shell/trunnion junctions. That is, the shell stiffness is not very much greater than the weld junction stiffness. For this reason the NRC staff finds that the bending stresses in the vicinity of the shell/trunnion junction should be evaluated, as required by ANSI N14.6. However, PSN used a STIF 42 element to model the local stresses. The STIF 42 element has only two degrees of freedom at each node and therefore cannot predict bending effects. Thus, owing to the type of finite element chosen by PSN, and the load model which precluded evaluation of bending, the analysis is inconclusive.

   PSN is requested to evaluate the bending stresses on the inner and outer surfaces of the outer shell in the vicinity of the lifting trunnion to prove that the ANSI N14.6 guidelines for critical lift safety factors are met.

2. PSN has supplied a top cover plate (actually a ring) which is to be fitted to the top of the MTC following the welding of the shield lid and top lid of the MSB. The purpose of the MTC lid is to prevent radiation streaming. Since the location of the hoist rings of the MSB is a circle of smaller diameter than the inside diameter of the MTC lid, a sling can be attached to the MSB hoist rings. The crane operator could conceivably lift the MSB higher than necessary to clear the bottom of the MTC shield door assembly. Should the crane operator do this, the MTC top cover plate would effectively be a lifting device, and the load would be the weight of the MTC without the lid.

   PSN analyzed this case; however, they only showed that the lid will not yield or fail for safety factors which do not meet ANSI N14.6. Consequently, PSN is requested to design the lid such that it will not yield or fail according to the ANSI N14.6 safety factors of 6 and 10 respectively. (Since the MTC is balanced on top of the VCC, which is supported by the heavy haul trailer inside the spent fuel pool building, the NRC staff finds that critical lift safety factors are appropriate, i.e, 6 on yield and 10 on ultimate.) Please show that the combined shear and tensile stresses at the critical sections meet the above safety factors.

3. The top cover plate of the MTC is secured by sixteen bolts. For the case of the inadvertent lifting of the MTC, as described in the above paragraphs, the bolts will take out the load and are therefore also, like the cover plate, effectively lifting devices. These bolts were analyzed by PSN. The NRC staff found that PSN used the gross cross section of the bolts instead of the tensile cross section, as required by the AISC. Furthermore, they compared the tensile stresses in the bolts to ultimate strength of the material, whereas the AISC code is very specific that the tensile stress should be 20.0 ksi for A 307 bolts. Consequently, although the bolts may not fail, they do not meet the AISC allowable stresses. They also do not meet the ANSI N14.6 safety factors.

The NRC staff finds that the bolts securing the top cover plate of the MTC shall meet the critical lift safety factors, i.e., 6 on yield and 10 on ultimate strength. Please provide analyses and design modifications to show that ANSI N14.6 requirements are met.

II. Specification and/or Calculational Comments

1. Based on the methodology recommended in Reference 18, PSN estimated the nil ductility transition (NDT) temperature of the steel. In order to start the process of NUREG/CR-1815, PSN calculated the minimum temperature of the MSB on a day for the coldest ambient temperature required by 10 CFR 71 after storing the fuel for twenty years. The lowest ambient temperature used as -20°F. The lowest temperature of the MSB was found to be 2.5°F, and by using Figure 3 (Category I) of NUREG/CR-1815, PSN determined that the NDT temperature is -32.5°F. The NRC staff concurs with this value, however subsequent values determined by PSN do not appear to conform with paragraph 5.1.1 of Reference 18 insofar as the determination of the minimum dynamic fracture toughness, K_{ID} and the minimum Charpy V-notch test value. The NRC staff determined that a minimum value for absorbed energy of 25 ft.-lb. is required to meet the guidelines specified in NUREG/CR-1815. This minimum absorbed energy value should be obtainable from Charpy V-notch specimens tested at a temperature of -30°F, which corresponds to the NDT temperature (-32.5°F) of the steel.

   The NRC staff finds that the Charpy V-notch value specified in the Appendix 2.2 of the TR Revision 3 p. 11, should be changed from 12.6 ft-lb. to 25 ft-lb. of absorbed energy measured at -30°F. Please make this change in specification.

2. The discussion of the accident pressure case for the MSB should be changed in the TR, to correspond with the actual worst case scenario of the MSB inside of the MTC. The resulting pressure is 32.6 psig instead of 28.6 psig. All pressure stresses in the MSB which are shown in Table 11.2-3 should be changed to reflect the increased pressure.

3. Heat transfer through the region above and below the fuel was modeled as conduction only with a conductivity of 0.1 BTU/hr-ft-°F. This is conservative in that it underestimates the heat transfer out of the top and bottom of the MSB and results in calculation of high fuel temperatures. With this approach a large temperature difference is predicted between the fuel and the MSB top and bottom. As a consequence the temperature gradient in the MSB top and bottom surface is underestimated. Although conservative for the purposes of calculating maximum fuel clad temperature, the method is not conservative for predicting thermal stresses in the MSB top and bottom.

   PSN is requested to re-evaluate thermal stresses in the MSB bottom. The NRC staff estimated the differential temperature to be approximately 310°F measured between the center of the MSB and the shell. See nodes 81 and 83 for the ambient temperature case of -40°F. (Figure 4.4-2 of the TR).

4. PSN did not submit stresses at top and bottom surfaces of the storage sleeve, therefore no bending stresses were available to the NRC staff to review. The bending stresses are required for evaluation, see Figure XIII-1141-4 of Appendices of the ASME code.

5. The manufacturer of the hoist lift ring specified a factor of safety of 5 on ultimate loading rating, but did not cite any factor 1 on yield strength. In order to meet NUREG-0612, the yield strength safety factor must be specified.

6. Since the MTC is fabricated from ferritic steels, PSN should cite what the minimum operating temperature of the MTC shall be in order not to risk brittle fracture.

7. In the design criteria section and/or Section 12 of the TR, PSN should note a minimum operating temperature for the MSB in order not to risk brittle fracture.

8. The general corrosion rate of carbon steel is affected by the temperature, humidity, and atmospheric pollutants, including chlorides, sulfides, etc. Values for the long-term corrosion rate of carbon steels in semi-industrial or moderate marine environments range up to 0.002 inches per year (Metals Handbook, Volume 13, Ninth Edition p. 532, September 1987). Rates in steam environments are about 0.012 inches per year. Severe marine environments yield corrosion rates up to about 0.018 inches per year.

   Thus, for semi-industrial or moderate marine environments under any expected condition, a value of 0.012 inches per year should be used. The predicted wall thickness affected for the expected 20-year storage period would be 0.24 inches. (This could be reduced as temperature drops below 212 in VCC.) This leaves an unaffected thickness of 0.51 inches. The above prediction of the effects of general corrosion are made by the NRC staff. In design calculations presented by PSN, a less conservative estimate of reduced wall thickness was made based on 0.003 inch per year and no steam, for a period of 50 years. Because the VCC has operating temperatures which exceed 212°F for ambient temperatures above 100°F with solar load (Table 4.1-1 of TR), there will be steam in the annulus between the VCC and MSB. The MSB is to be coated with a protective coating specified in the fabrication specification. PSN did not take any credit for whatever degree of protection the coating might provide. The evaluation of stresses due to the drop of a corroded MSB were presented in the same TR design calculation. Membrane stresses were obtained by multiplying the drop stresses by the ratio of non-corroded thickness to the corroded thickness. Bending stresses were obtained in a similar manner except that the ratio of the squares of the thicknesses were used. For the horizontal drop case shell stresses are 71.8 ksi. With normal internal pressure and dead weight, as required by ASME code load combination for Service Level D, the stress is 73 ksi. The Code allowable is 73.5 ksi. Therefore, for the drop condition, the MSB design has virtually no allowance for any corrosion in the shell.

Consequently, PSN needs to increase the wall thickness of the shell to account for general corrosion.

9. Table 3.4-6 of the TR evaluates 8 load combinations, one of which is equation 6, which has an accident term. The TR table and a design calculation submitted by PSN shows that the VCC will withstand a 22 g horizontal drop deceleration. The NRC staff evaluated the PSN submittal and has come to the conclusion that the calculation presented by PSN is incorrect. Furthermore, by using a conservative method, the NRC staff calculates that the load combination number 6 (ANSI 57.9) results in a moment capacity far greater than the allowable moment capacity.

   PSN neglected to use units of mass in the calculation for the natural frequency of the VCC. If mass units had been used a much higher frequency and a much higher value for the required moment would have resulted. The required moment will exceed the allowable moment. Therefore, the VCC does not meet the design criteria. However, the NRC staff can accept the design based on a limiting condition of operation. If the VCC/MSB is dropped, the MSB will according to other analyses not fail. Thus, the confinement boundary will not be breached. The VCC may fail, but such a failure will not affect the MSB except possibly from thermal hydraulic considerations. In any event, the VCC/MSB drop accident will require recovery of the MSB and subsequent inspection of the MSB and the VCC. If the VCC is damaged beyond repair it must be removed from service.

   In summary, PSN may elect to recalculate this drop condition with the objective to show that the VCC design does meet equation 6 from ANSI 57.9, and therefore meets the stated design criteria. PSN may elect to state in the TR that the VCC design does not meet the design criteria, and using an argument suggested by the NRC staff above, specify limiting conditions of operation to be written in to Chapter 12 of Revision 3 of the TR. Procedures for recovery should also be adequately detailed.

10. A partial explanation of the residual dissatisfaction with the criticality safety evaluation in the SER is that the approach for criticality safety presented in the SAR is different from the basis for criticality safety that the NRC is current prepared to accept. The SAR presents a case for assurance of criticality safety based on burnup credit. Misloading of unirradiated fuel is discussed as an accident case. The basis for the criticality safety review for all the ISFSI SERs to date is that the design should provide assurance for criticality safety for the loading of unirradiated fuel assemblies into the storage canister under optimal water moderation conditions. For the NUHOMS-24 design, the safety criteria were modified to accommodate a design where there was no possibility of accidental flooding of the canister. If the loading and loading operations were conducted in such away that boiling could not occur, then we established a reactivity criterion of 0.98 for flooding during the loading and unloading operations. This was justified because the flooding operation would be closely monitored and controlled. If boiling cannot be precluded, then the 0.95 reactivity criterion for

optimal moderation would still apply. The NRC staff has tried to apply the same criticality safety criteria for the VSC-24 design. Since the VSC-24 SAR was not written to overlay with the criticality safety criteria, the staff has tried to fit what was presented in the SAR to the criticality safety criteria. It would have been helpful if the SAR were written to match the basis for the review. Based on criticality safety criteria, two remaining questions have been identified.

1. Figure 11.2-6 is an important result and new to Revision 2. Specific questions for this figure:

- What fuel design was used for the results presented?

- Do the results include uncertainties? Which uncertainties? Please show a breakdown of the constituent results in tabular form for the specific points that are plotted in the figure.

- Are the curves extrapolated to 5 percent enrichment or are they calculated results at 5 percent enrichment?

2. The question of boiling during loading and unloading operations has not been satisfactorily discussed. On what basis can boiling be dismissed? If the possibility of boiling cannot be precluded, then what is the minimum time after fuel assembly loading that boiling could commence? How is boiling controlled during refilling prior to unloading?

11. As part of the review process for the PSN shielding design, independent calculations were performed for the air scattering of neutrons from the loaded VSC to a distance of 50 and 1000 feet using the MORSE-PC computer code. MORSE-PC is a personal computer version of the MORSE monte carlo neutron/gamma transport code which has been used extensively in the nuclear industry for about 20 years. A three dimensional R-Theta-Z model of the VSC including the MSB and all associated internal components was developed. The PSN neutron source was applied to the outer surface of the MSB. This model utilized the CASK 22 neutron and 18 gamma ray energy group cross sections which were applied to the constituent materials in the VSC (i.e., concrete, steel, air, and RX-277 neutron shielding material). Russian roulette techniques were used to optimize computational resources.

The aforementioned MORSE-PC model of the PSN VSC-24 was utilized in calculating the neutron dose rate at a distance of 50 and 1000 feet from a single fully loaded VSC. The calculated neutron dose rate was a factor of 36 times higher than the PSN value at 50 feet from the single cask and a factor of 77 times higher than the PSN value at 1000 feet from the single cask. The PSN values were calculated using the SKYSHINE-II computer code.

The neutron dose rates at a distance of 50 and 1000 feet from a single VSC cask were calculated to be between 36 and 77 times higher than the values presented by PSN. Although the neutron dose rate is a small fraction (about 0.1 percent for the PSN calculated dose rate values) of the overall dose rate, this large disparity in calculated scattered neutron dose rate between MORSE-PC and PSN points to a possible similar difference for calculated gamma dose rates. The effect of increasing the scattered total dose by such factors would be to require a population exclusion radius in excess of 2000 feet. The difference in calculated scattered dose rates needs to be resolved to address this issue. Appropriate validation of the PSN scattered dose rate methodology with the SKYSHINE-II computer code should be performed. This validation should include benchmarking against experimental data for air scattered neutron and gamma sources and appropriate model-code sensitivity studies.

III. Comments on Drawings

1. Drawing MSB-24-001 Rev. 2, Sheet 2 of 2, Section D-D should be changed to correspond to Figure 3.4-1 of the TR. This change is necessary to achieve a double seal weld.

2. The lift ring manufacturer specified that no shim washers may be used under the bushing flange. Please add a note to drawing MSB-24-002, sheet 1 of 2 to this effect.