January 20, 2010

MEMORANDUM TO: Christine A. Lipa, Branch Chief
Division of Nuclear Materials Safety, Region III

FROM: Eric Benner, Branch Chief **/RA/**
Licensing Branch
Division of Spent Fuel Storage and Transportation

SUBJECT: RESPONSE TO REGION III TECHNICAL ASSISTANCE REQUEST – PALISADES PLANT MULTI-ASSEMBLY SEALED BASKET NO. 4 (MSB #4) - WELD FLAW ANALYSIS

By submitted Technical Assistance Request (TAR) dated November 01, 2006, the Division of Nuclear Materials Safety, Region III (DNMS, RIII), requested the Division of Spent Fuel Storage and Transportation (SFST) provide an assessment of issues originally concerning Consumers Power Company (previous licensee) and presently Entergy Nuclear (current licensee), Palisades Nuclear Power Plant, Multi-assembly Sealed Basket, No. 4 (MSB #4). DNMS, RIII, requested assessment of issues concerning previous licensee's engineering analysis of three indications identified in one of the confinement boundary welds. The engineering analysis evaluated the structural integrity of MSB #4 which was loaded with spent fuel in July 1994 and is currently stored onsite, Palisades Plant.

SFST staff completed its review of the engineering analysis concerning MSB #4 weld flaws and provides the enclosed assessment in response to your TAR. SFST staff concludes that the methodology with assumptions, results and conclusions documented by both previous and current licensee and its consultants to be acceptable.

CONTACT: David Tarantino, SFST
301-492-3413

Docket No: 50-255; 72-007.

Enclosure: DSFST Ticket No. 200700005, Response to Region III TAR – Palisades Plant MSB #4 - Weld Flaw Analysis

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Response to Region III Technical Assistance Request (TAR)
For Palisades Nuclear Power Plant
Multi-assembly Sealed Basket No. 4
DSFST Ticket Number: 200700005
Prepared By: David M. Tarantino

REQUEST:

By submitted Technical Assistance Request dated November 01, 2006, the Division of Nuclear Materials Safety, Region III (DNMS, RIII), requested the Division of Spent Fuel Storage and Transportation provide an assessment of issues originally concerning Consumers Power Company (previous licensee) and presently Entergy Nuclear (current licensee), Palisades Nuclear Power Plant Multi-assembly Sealed Basket, No. 4 (MSB #4).

The previous licensee performed an engineering analysis (EA) of three indications in one of the confinement boundary welds to evaluate the structural integrity for interim operability of MSB #4, loaded with spent fuel in July 1994 and currently stored on the Palisades Site. DNMS, RIII, requested an assessment of previous licensee's EA concerning the following issues:

- The characterization of the weld indications and of the root cause.
- The adequacy/acceptability of assumptions, methodology and conclusions, which demonstrated that MSB #4 is structurally sound and meets its design basis.
- The adequacy/acceptability of assumptions, methodology and conclusions, which demonstrated that MSB #4 is leak tight and the radiation exposure to the public meets regulatory requirements.
- The agency's previous records and decisions regarding the operability of MSB #4 and the licensee's plans associated with MSB #4.

BACKGROUND:

In April, 1992, Metals Evaluation and Testing, Inc. (MET) performed the original Radiographic Test (RT) inspections on MSB #4 as contracted by fabricator Richmond-Rhodes Enterprises. No crack-like indications were identified in the longitudinal seam (LS-1) weld.

In 1994, previous licensee had already loaded spent nuclear fuel (SNF) into MSB #4, a component of the Ventilated Storage Cask No. 24 (VSC-24) System, when concerns over implementation of a vendor's Quality Assurance program prompted a second or re-review of RT reports and associated films for MSB #4. During this second review a licensee level III inspector identified an apparent indication (elongated crack-like indication, 1-inch long) in weld LS-1. Consequently, all RT reports and associated films for MSB #4 were subjected to a third review performed by a licensee level III inspector and an independent level III inspector from PCI Energy Services. Both level III inspectors identified three indications in the MSB #4 shell weld LS-1 as follows: a longitudinal crack-like 3/4-inch long, at location 52, a transverse crack-like 5/16-inch long, at location 57 and a linear slag-like 3/8-inch long at location 116.

EVALUATION:

Previous Licensee Analysis, Assumptions, Conclusions, Root Cause and Actions:

The licensee conducted an engineering analysis (September, 1994) of structural integrity to determine the magnitude of flaw propagation subject to fatigue loads throughout the 50-year design life of MSB #4. The flaw propagation results were then used to evaluate the stability of the flaw size under normal and faulted load conditions. In addition, the licensee concluded the cracks were not through-wall. This conclusion was based on the MSB #4 having been sealed, vacuum dried and back-filled with helium. The MSB #4 was leak tested, alternating between pressurizing (water/helium) and pulling a vacuum on the MSB internals, followed with a final backfill of helium pressurized to 14.5 psia.

The potential for crack propagation or material failure were computed by Sierra Nuclear Corp. (SNC) and Structural Integrity Associates (SIA), who were contracted to perform a conservative stress and fatigue analysis. ASME Code, Section XI, 1992 Edition, guidelines were applied with the following assumptions:

- Negligible thermal stresses are expected based on design heat load and shell heat conductivity (~1ºF/inch drop across (inside-to-outside) and axial (within) the MSB #4 shell).
- No movements or forces transmitted from the surroundings are expected since the MSB #4 is loaded unrestricted within the Ventilated Concrete Cask (VCC).
- No internal forces on the MSB #4 shell are expected due to the storage sleeve stress-free geometry.
- Small hoop stresses from variations in internal pressure and compressive loads due to shell and lid weight are expected.
- The hypothetical accident condition (HAC) (fuel rods rupture, release fission products and fill gas) was utilized to produce an internal MSB pressure increase up to approximately 35 psig for use in the hoop stress fatigue (cycling) analysis. Pressures of 1,200 psi were applied to the shell through 20,000 cycles, which is derived from conservatively applying 1 cycle/day over 50-years.
- A cylinder through-wall axial crack was postulated and used in a linear elastic fracture mechanics (LEFM) model for flaw evaluation.
- A Charpy V-notch impact energy of 12 ft-lbs was assumed to calculate fracture toughness utilizing a correlation equation provided by Welding Research Council Bulletin 265 (February, 1981). The bulletin states that correlations between fracture toughness (FT) and small-scale test results, such as Charpy V-notch, are useful in pressure vessel applications due to cost, availability of material, and ease of testing.

SIA calculated FT using the correlation equation between Charpy V-notch (CVN) and FT as follows: $K_{Ic}=[2E(CVN)^{1.5}]^{½}$, where FT parameter ($K_{Ic}$) is stress intensity factor, Young's Modulus (E) of the shell was set to 30 x 10^6 psi and CVN was assumed to be 12 ft-lbs. The calculation resulted in a safety factor (SF) of 38 against failure. In addition, a fatigue crack growth calculation for ferritic steel in an air environment resulted in minimal growth over the 50-year cask design life. The licensee reviewed SIA's calculations and concurred with SNC conclusions.

The licensee calculated minimum wall thickness required by the ASME Code to be 0.31-inches for the MSB #4 shell under normal operating conditions (NOC). The licensee compared the actual shell thickness and calculated the minimum wall thickness required for NOC to demonstrate the design margin of the MSB #4. The VSC-24 safety analysis report

(PSN-91-001, R1, October, 1991) provided design parameters used with ASME Code, Section III, NC, 1986 Edition design requirements and acceptance criteria to evaluate dead weight, thermal, pressure and handling loads during NOC. A uniform shell thickness was assumed and the largest ratio of calculated stress to ASME Code stress limit (primary local membrane plus primary bending stress) was equal to a stress margin safety factor of 8.3.

The analysis consisted of three core calculations for the minimum wall thickness required under loads associated with internal pressure, normal operating stress or loading conditions and external pressure. Fatigue is not required in this analysis as the total cycles (loading) are acceptable under ASME Code requirements as determined by the VSC-24 design safety analysis. The MSB #4 design loading includes the following:

- Internal pressure (0.7 psig max) varied with outside ambient temperature (125°F max) associated with the off-normal condition, was used for conservatism and generated both membrane and bending stresses as a result of local shell distortion between the bottom and head connections.
- Dead Weight (structural plus shielding lid plus shell equaling 16,140 lbs.) generated a membrane stress.
- Thermal (stress caused by axial temperature gradients that expand shell top more radially than shell bottom) generated a bending stress proportional to the shell stiffness.
- Handling load (caused by movement/handling, normal handling is $\pm$0.5 g's acceleration) was proportional to shell mass.

Further, minimum wall thickness was examined due to the impact of the following accident condition loads: dead weight, pressure, thermal, handling, drop, seismic, flood and tornado loads. The following associated assumptions were made: the normal handling load will be used in lieu of seismic load (which is greater and therefore conservative); the postulated tornado loads will have no effect on (overturning) the MSB #4 enclosed within the VCC; flood loads (buoyancy, static and velocity pressures) are not considered adequate to elevate or tip the VCC nor submerge the cask due to its elevation; no drop or handling loads are considered valid as the MSB #4 will not be transported; a maximum internal pressure of 34.6 psig is expected as a result of the fuel rod HAC defined above (resulting in a calculated 0.022-inch minimum wall thickness) and the dead weight, thermal and handling load additions were neglected.

The calculated MSB #4 minimum wall thickness required by ASME Code for NOC is 0.31-inches based on analysis using the conditions described above. However, this minimum wall thickness is a combination of weight, pressure and handling membrane plus bending stresses. No further handling of the MSB #4 is expected, therefore, removing the stress produced from the handling design load results in a required minimum wall thickness of 0.193-inches as compared to the MSB #4 design nominal wall thickness of 1-inch.

In addition to helium leak testing, the licensee performed increased radiological monitoring of VSC #4 and the surrounding storage pad area. No unusual dose rates were recorded and contamination was not detected. Daily (August, 1994 through August, 1995) smear contamination surveys on all four VSC #4 outlets were performed in addition to measuring the dose rates on all four VSC #4 inlets and outlets.

Conclusion/Root Cause/Corrective Action:

The licensee concluded that the RT film containing both crack-like indications identified at locations 52 and 57 was probably missed during the original RT interpretation. Also, the slag-like indication at location 116 is at the acceptance criteria limit and was apparently accepted. In addition, the licensee considers the two crack-like indications to be cracks, as no further testing is practical and the one slag-like indication is at the limit of the acceptance criteria, which is 1/3 of the weld thickness (i.e., 1/3 of 1-inch).

The licensee determined that of the two dissimilar conditions identified in the MSB #4 LS-1 weld a single root cause exists. Without a second party review of the radiographs there was no way to confirm the acceptance of the slag-like indication or to have reduced the chances of missing the RT film containing both cracks.

As a result, the licensee revised the MSB fabrication process to require a hold for independent review of the radiographs and to develop a plan for addressing the long-term resolution of fuel assembly storage in MSB #4.

NRC Staff Assessment (Previous and Current Licensee):

The previous licensee concluded the crack-like indications originated from a cracked tack weld not completely removed while back-gouging or completely consumed during welding of the outside diameter groove and an unremoved end of a weld-crater crack. In addition, the cracks, both of which were located on the same piece of RT film, were likely missed during film evaluation and the slag, bordering on the ASME Code criteria limit, was likely accepted. Further, the root cause was established as lack of a second independent party review performed during the original RT film reading.

The staff concludes that the origin and cause of the missed indications are indeterminate. Insufficient data exists to formulate a distinct conclusion as general observations from information provided could suggest other reasonable explanations. Slag, porosity and cracks are all common defects caused by the weld process used on the MSB #4 shell. For example, slag and porosity are the only defects recorded on the MET radiography reports (excluding LS-1). Slag, occurring only in weld metal, can be ascribed to technique, type and configuration of weld joint and may occasionally appear as transverse cracks. Properly preparing the weld surface can reduce the risk of slag inclusions and may include grinding and/or air arc-gouging. The staff notes this is also part of the licensee's basis for concluding the origin of the cracks (i.e., incomplete back-gouge and improper preparation of subsequent weld surfaces). In addition, the staff notes that tack welds used to secure alignment shall either be removed completely, when they have served their purpose, or their stopping and starting ends shall be properly prepared by grinding or other suitable means so that they may be satisfactorily incorporated into the final weld. Tack welds should only be made by qualified welders using qualified welding procedures. When tack welds become part of the finished weld, they should be visually examined and defective tack welds should be removed.

The staff recognizes that leak tightness was confirmed based on the vacuum drying and helium backfill portion of the leak tightness evaluation. During loading and placing of the VSC #4 into storage no leaks into or from the MSB #4 were detected. The leak tightness test begins by pressurizing the MSB with water and holding the pressure within a range for a predetermined amount of time. The MSB is then partially drained of water and pressurized with helium. Next, a vacuum is drawn on the MSB in a series of increments until a final vacuum pressure is maintained and verified. The vacuum is isolated and the MSB is again pressurized with helium while lid welds are checked for leakage. The helium is vented subsequent to a satisfactory leak check and a final vacuum is drawn on the MSB as described above. Lastly, the vacuum is isolated and a final backfill of helium pressurizes the MSB to a pressure of 14.5 psia for storage. Further, in-service accelerated radiological monitoring of the VSC #4 was performed as part of the leak tightness evaluation upon placing the VSC #4 into storage. Daily contamination and dose rate measurements were taken to determine if contamination was present or an above-normal dose was recorded. The radiation survey results showed no contamination or uncharacteristic radiation dose was present at VSC #4 and the surrounding storage area.

The staff notes that a fracture mechanics (flaw tolerance) analysis was performed to determine whether the structural integrity of the MSB #4 LS-1 weld was compromised due to the presence of the flaws, subjected to the various design loading conditions. The analysis was performed, with assumptions, to determine whether a postulated flaw was stable under both static and fatigue (cyclic) loading conditions. The LEFM method of examination is conservative as plastic contributions of the material to resist or arrest crack propagation are neglected. In addition, the following assumptions were used to further establish a conservative flaw model. A crack direction normal to the applied stress was assumed resulting in the largest stress intensity factor as compared to other crack directions. A single V groove joint design was assumed (in lieu of double V groove) providing a larger volume of weld metal, thus producing an increased residual stress. A surface connected through-wall flaw was assumed (in lieu of subsurface flaw) increasing the stress intensity factor. And flat plate geometry was assumed neglecting the small curvature effects of the MSB #4 shell. Staff conclude that the fatigue crack growth would be negligible for the 50-year design life of MSB #4 based on the LEFM analysis and the conservatively assumed loads. Stress levels in the MSB #4 shell are low and inadequate to cause significant crack growth. In addition, the crack remains stable when subjected to accident conditions and will not grow through-wall as a result of fatigue or one-time accident loads.

Staff review of the analysis performed by the previous licensee noted the base material yield stress was used to assume a uniform welding residual stress. The primary driving force for fatigue crack growth or the stress intensity range is not affected by residual stress. However, the stress intensity ratio, which is a secondary variable used to ascertain fatigue crack growth rate is affected. The stress parameter (R) used in calculating fatigue crack growth rate, assumed a possible non-conservative constant loading cycle value of R = 0.9. The resulting fatigue crack growth rates would therefore also be non-conservative and the results could under-predict the eventual flaw size.

The staff requested the value of R be increased to 1.0. The current licensee performed a reassessment to determine the flaw size at the end of a 50-year design life using an R value of 1.0. The methodology and other assumptions remained unchanged. The fatigue crack growth results remained essentially the same as the original analysis. Both the normal (4.93) and faulted (2.97) condition flaw stability safety margins are greater than the required safety factors of 3.16 and 1.414, respectively. The staff finds the methodology, assumptions, and conclusions provided to be acceptable.

STAFF CONCLUSION:

Re-evaluation of the RT film detected one slag-like and two crack-like indications in the longitudinal seam (LS-1) weld where no indications had been reported during the original evaluation. The licensee decided to characterize the indications and evaluate the structural integrity of MSB #4 by performing a flaw propagation and stability analysis in lieu of a weld repair.

Staff conclude the analyses provided by the licensee demonstrates the ability of the MSB #4 cask to perform its intended function for the 50-year design life. Calculations provided by the licensee and confirmed by staff demonstrate that the MSB #4 shell material has a considerable flaw-tolerance margin. In addition, the calculations demonstrate that the flaws will remain static during in-service operation and will not propagate as a result of any design basis normal or off-normal events. Staff conclude that the MSB #4 is leak-tight in its present condition and that MSB#4 will not pose an increased radiological impact to the public or environment based upon information provided by the licensee.

REFERENCES:

1) Consumers Power, Palisades Nuclear Plant Engineering Analysis, EA-FC-864-050, "MSB #4 Structural Integrity Assessment."

   - Attachment 1, Consumers Power Company, Internal Correspondence, JCN94 * 031, Dated August 22, 1994, "Postulated causes for MSB #4 flaws."

   - Attachment 2, SIERRA NUCLEAR CORPORATION, Dated August 15, 1994, SNC-94-394, "Analysis of Longitudinal Weld Crack on MSB #4."

2) Consumers Power, Palisades Nuclear Plant, Engineering Analysis, EA-C-PAL-94-0617-01, "Minimum Required Wall Thickness for the MSB Shell During Normal Conditions." (Appendix 1 to EA-FC-864-050).

3) Consumers Power, Palisades Nuclear Plant Engineering Analysis, EA-FC-864-050, "Appendix 2 to MSB #4 Structure Integrity Assessment."

   - Attachment to EA-FC-864-050 Appendix 2, Sheet 1&2/6, Alloy Rods Corporation Certificate of Analysis, "Certified Materials Test Report," Dated 04-04-1991.

   - Attachment to EA-FC-864-050 Appendix 2, Sheet 3&4/6 Testing Engineers, Inc., "QM-483, Suggested Format for Procedure Qualification Record (PQR)," Dated 01-20-1992.

   - Attachment to EA-FC-864-050 Appendix 2, Sheet 5/6 Testing Engineers, Inc. "Charpy V-Notch Impact Test," Laboratory Number: M1061, Dated 01-17-92.

   - Attachment to EA-FC-864-050 Appendix 2, Sheet 6/6, Testing Engineers, Inc., "Mechanical Tests," Work Request Number M1031, Dated 01-16-92.

4) Consumers Power, Palisades Nuclear Plant Engineering Analysis, EA-FC-864-050 Appendix 3, "MSB #4 Leak Tightness Evaluation."

- Attachment 1 to EA-FC-864-050 Appendix 3, "RADIOLOGICAL SURVEY SHEET," RWP 940333, Item: Attachment 7A, Holdpoint #13, Dated 07-16-94.

5) Consumers Power, Palisades Nuclear Plant Engineering Analysis, EA-FC-864-050 Appendix 4, "MSB SHIELDING DOSE EXPOSURE EVALUATION."

6) Sargent & Lundy, Entergy, Palisades, Project No. 12122-035, Calc. No. 2007-20168, Rev. 00, Dated 12/12/07, "EA-FC-864-050, Palisades Weld Flaw Analysis for loaded VSC Spent Fuel Cask MSB No. 4."

7) Welding Research Council Bulletin, BULLETIN 265, "INTERPRETIVE REPORT ON SMALL SCALE TEST CORRELATIONS WITH K_{Ic} DATA," Dated February 1981.

8) NUREG/CR-1815, "Recommendations for Protecting Against failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick."

9) Palisades Nuclear Plant Permanent Maintenance Procedure, Proc. No. FHS-M-32, Revision 5, Pages 32 thru 43 of 65, "LOADING AND PLACING THE VSC INTO STORAGE," Dated 07-13-94 thru 07-15-94.

10) "Safety Analysis Report for the Ventilated Storage Cask System," Rev. 1, PSN-88-018, Dated October, 1991.

11) American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1992 Edition thru 1994 Addenda.

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