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MONTICELLO NUCLEAR GENERATING PLANT (DOCKET NO. 50-263) - REACTOR VESSEL

We have completed our review of the reactor vessel for the Monticello Nuclear Generating Plant. Our report is enclosed.

/S/

A. W. Dromerick, Chief
Containment & Component Technology Branch
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Enclosure:
Report on FSAR Review of Monticello Plant

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ACRS REPORT

MONTICELLO REACTOR VESSEL

1.0 INTRODUCTION

1.1 General

Serious problems arise with respect to the transportation of large power plant components to remote site locations for which access is limited to secondary roads and railroad tracks. The Monticello site is such a location. For this plant, it was decided to pre-fabricate the reactor vessel subcomponents to the maximum degree compatible with the transportation limitations and to complete the vessel fabrication at the site. The ASME Code, Section III, does not require vessels to be fully fabricated in one place such as an established shop. However, because it was a major departure from conventional practice, our evaluation of the vessel has been conducted to a somewhat greater depth.

1.4 Status of Reactor Vessel

The reactor vessel fabrication, including the CRD housing field welds, has been completed by CB&I. The ASME Code required hydrostatic test is completed and the Code Stamp has been affixed.

2.0 REACTOR PRESSURE VESSEL

2.1 General

The Monticello reactor vessel is designed and fabricated in accordance with the 1965 edition of the ASME Section III, Class A, Code plus 1966 Addenda, and Code Cases 1332, 1335, 1336, and 1339.

The vessel core support structures are designed to accommodate jet pumps, and the internal support structure is made of Inconel and stainless steel which has not become furnace sensitized during the fabrication cycle. CRD (control rod drive) stub tubes are made of Inconel, and the distance between the shop weld and the field weld (to CRD housing) is approximately 4 inches. It should be recalled that this distance was a minimum of 0.5 inch for Oyster Creek and Nine Mile Point stub tubes and this, together with the stub tubes being of stainless steel, resulted in high stub tube stresses during operation. The stub tubes are attached to the bottom head in counterbores with a partial penetration weld, a Section III approved method.

The reactor vessel is made of low alloy carbon steel type A-533 Cl.1, Grade B. The vessel interior is clad with Type 308 stainless steel applied by weld overlay technique.

The Monticello vessel has the same geometry as specified for other BWR type vessels except for the recirculation inlet nozzles. These nozzles are located at a 9-inch higher elevation than for similar vessels. This feature was necessary in order to accommodate the field welding of a girth seam without incurring concernable weld distortion in the nozzle vicinity.

Thermocouples are attached to the vessel starting at the closure region and continuing down along the side of the vessel to the support skirt region. These thermocouples are connected to a panel in the main control room where the temperatures will be observed and recorded during the initial run of operational transients. The observed temperatures will be used to verify assumptions for the reactor vessel transient stress analysis.

2.2 Design Specification

The design specification is basically identical to specifications used for reactor vessels for plants such as Dresden 2 and 3 and Millstone. Specific differences in the specifications are associated with special provisions related to items such as electroslag welding (not used for this vessel), and separation of responsibilities.

The FSAR specified 50,000 cycles of Rod Worth Tests for thermal loading. This transient was interpreted at a meeting with the applicant on October 2, 1969, to consist of: 50,000 cycles of 10 °F temperature changes, and 400 cycles of 30 °F temperature changes. Because a 10 °F change in water temperature has an insignificant effect on the vessel stress level, this transient is not analyzed. GE's standard equipment specification for future vessels will list only the 400 cycles of 30 °F. Amendment No. 21 to the FSAR, dated 10/17/69, includes a corrected Table 4-2-1, REACTOR VESSEL ANALYSIS, with the Control Rod Worth Tests reduced to 400 cycles.

The staff concurs that omission of the 50,000 10 °F cycles does not influence the vessel stress level and the cumulative fatigue usage factor in a significant manner.

Also included in the specification are inspection requirements over and above the Section III required. Notably among these are 100% ultrasonic testing (UT) of plate material, UT of cladding bond on a grid pattern, Charpy V-notch toughness testing with enough samples to obtain a full Charpy curve including upper and lower plateaus, and nil ductility transition temperature (NDTT) determination by Drop Weight testing.

Since CB&I did not initially have a stress analysis group of sufficient size and experience, this task was divided between General Electric (GE) and CB&I. GE would perform the transient thermal analyses and the cyclic operation analyses for the components. In addition, the applicant agreed to provide an independent review of the stress analyses.

2.3 Stress Analyses

The results of the stress analyses show that the highest cyclic stresses will be experienced by nozzles for recirculation inlet, feedwater, CRD hydraulic return, and core spray and flooding. All of these nozzles are provided with thermal sleeves to reduce the stress levels to within code acceptable limits. High stresses are also indicated for the vessel support skirt near the vessel bottom. Stress levels for CRD stub tubes are low enough to permit shakedown to elastic stress levels after the initial operational cycles.

All calculated stress levels conform to the limits of Section III.

Cumulative usage factors have been calculated to be 0.52 for the core spray nozzle, 0.4 for support skirt attachment, and 0.56 for the closure stud bolts. Other calculated usage factors are less than 0.1.

Amendment No. 21, dated 10/17/69, referenced GE Topical Report APED-5703 on CRD penetration thermal and stress analyses. While we have not yet made a detailed review of this report, our preliminary review indicates that the design assumptions agree with the GE reactor vessel specification.

The Teledyne Materials Research Division, acting for the applicant, made an independent review of the stress analysis report and found it consistent with industry practice with respect to the analytical methods used and the ASME code interpretations employed.

Based on our review of the summary stress analysis provided by the applicant and the independent review by Teledyne, we have concluded that the design of the reactor vessel will provide adequate margins of safety.

2.4 Fabrication Details

Fabrication of the reactor vessel proceeded without any unusual problems arising. During stainless steel cladding of base metals some areas were found to develop microfissuring. Examination of the problem disclosed a ferrite content of 1-2%, while practice has established a 5-10% range as being normal to prevent cracking. Backtracking, it was found that a human error in calculating the ferrite percent in the qualification procedure had prevented corrective adjustment of the flux composition. Remedial action was undertaken by grinding off the affected areas to remove the surface crack indications, and in areas where the cladding thickness was reduced to less than 5/16 inch additional stainless steel was redeposited.

Although not considered a problem the fabrication procedure for the vessel differed from conventional practice in two ways:

1. The flat base plate material was Q & T (quenched and tempered) by the steel supplier rather than by the fabricator.
2. The plates were cold formed and stress relieved by the fabricator instead of the conventional hot forming followed by a Q & T heat treatment. The cold forming was performed after the plates had received sufficient preheat, 100 °F min., to move the material out of the transition temperature range and into the ductile range. The total forming strain was approximately 2-1/2%.

Experimental work has been performed to predict the degree of influence of cold forming or processing on plate materials,1/ however, most attention has been given to uniform strain and relatively little to the non-uniform strain conditions which occur during bending operations. ASME Section III considers cold forming effects in paragraph N-515, except that it does not place a specific limit on the amount of cold forming, provided the tensile and impact properties of the parts are not reduced below the minimum specification values. Essentially this means that the manufacturer must produce a procedure qualification test for forming operations which are not followed by an austenitizing heat treatment.

CB&I performed cold forming tests on A302-B and A533-B, Class I, materials. Total strain selected for these tests was 4% which is in excess of actual forming strain. Results of these tests show that:

1/ W. T. Lankford, "Effect of Cold Work on the Mechanical Properties of Pressure Vessel Steels." The Welding Journal, April 1956, pp. 195-S to 206-S.

a. Cold deformation of 4% stretch and subsequent stress relief does not significantly affect tensile properties of the plates at surface and 1/4 T levels. Surface tensile strength was reduced approximately 2000 psi and 1/4 T tensile strength approximately 1000 psi. However, the plate center tensile properties improved approximately 1500-2000 psi due to the cold working.

b. Impact properties were slightly improved at the surface levels, and they were considerably enhanced at the plate center.

c. Limited DW (drop weight) test information was obtained on the A533-B material. NDT temperatures varied from +10 °F to +50 °F for these tests.

We have concluded that for the Monticello vessel cold forming of the reactor vessel plates did not produce a concernable degradation of materials properties.

Drop weight test results to determine NDTT for the vessel shell and dome material ranged from +10 °F to +50 °F, except for one piece at 0° F. These test results are from the 1/4 T (thickness) level at the tensile stretch side (outside) and they represent the more conservative location, 1/4 T versus 3/4 T, for the samples.

Based on the results of the drop weight tests, which are better NDTT indicators, we intend to require that the vessel pressurization temperature be 110 °F instead of the 100 °F proposed by the applicant.

2.5 Dimensional Variations

The fabrication sequence did not result in any dimensional variances which would compel code consideration or introduce difficulties with fitting the internal structures into the vessel. However, the final overall measurements resulted in a 15/16 inch longer vessel and one inch larger inside diameter than called for by the specification. The effect is an insignificant increase in the vessel steam space and approximately 145 cu. ft. of additional water space, which is less than 1% increase of volume below normal water level.

It is noteworthy to observe that local vessel shell distortions at weld joints are less for this reactor vessel than for shop welded vessels. This condition is attributable to the weld groove geometry as compared to the deep j-groove geometry employed for automatic shop welding. The latter approach results in local inward bulging at the weld location, a condition which can be of concern in vessel areas with high coolant flow velocities.

It is also noteworthy that the dimensional location and verticality variations for bottom head CRD penetrations and stub tubes are all well within the specified tolerance range and, in fact, within a band equal to 20% of that range. This dimensional control was achieved through the use of shop produced boring templates, which eliminated the need for careful location measurements at the site once the templates were properly aligned and anchored.

We have concluded that the increased vessel dimensions are of insignificant magnitude.

2.6 Material Problems

Except for the CRD stub tubes, no concernable problems arose about the materials for the reactor vessel.

While machining inside and outside diameters of the CRD wrought Inconel stub tubes, axially aligned inclusions were noticed. These inclusions were of a type not readily detectable by conventional non-destructive inspection methods, and were found to exist on a limited number of stub tubes which were then rejected. Investigations and laboratory tests performed by CB&I, Illinois Institute of Technology Research Institute and others have disclosed the inclusions to be magnesium oxide. The Inconel 600 material is a flux-cast material and magnesium oxide is a constituent of the flux; other potential sources of magnesium oxide are magnesium (added as a malleabilizer and deoxidizer) and the firebrick of the mold, which under adverse pouring condition could increase the propensity toward slag entrapment. Although tests at CB&I showed the small inclusions to be innocuous, the use of any inclusion-bearing material was avoided. The Monticello reactor vessel consequently does not contain any detectable inclusions.

2.7 Sensitization of Stainless Steel

Aside from prudent material selections, the fabrication procedure and sequence of operations were established to reduce to a minimum the number of cases where stainless steel component parts of the vessel would become furnace sensitized. As a result only the following items are in a fully sensitized condition:

1. Weld deposited cladding overlay on vessel plates and forging.
2. Two recirculation outlet nozzle safe ends.
3. Two jet pump instrumentation nozzle safe ends.
4. The CRD hydraulic system return nozzle safe end.
5. Two top head instrument nozzle safe ends.
6. The top head vent safe end.

All of these vessel areas not available for inspection from the outside.

3.0 CONCLUSION

On the basis of our review, we have concluded that the Monticello field fabricated reactor vessel is acceptable for service.