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NUCLEAR REGULATORY COMMISSION
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REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

1.0 Introduction and Background

The NRR staff has reviewed a number of aspects of Millstone 2's recovery from the thermal shield damage discovered on June 30, 1983. This included two meetings with the staff as discussed in References 1 and 2. Our review included work by the Core Performance, Mechanical Engineering, and Materials Engineering Branches.

Reference 3 is the licensee's report in connection with the thermal shield removal and plant recovery program. The report contains a descriptive summary and chronology of events and a description of the reactor internals related to the thermal shield (Chapter 1 and 2). The parameters and values related to the Pressurized Thermal Shock (PTS) are summarized in Chapter 3. The calculation of the neutron fluence, the material properties and the pressure vessel energy deposition rate are discussed in this chapter. The non-destructive examination techniques applied to the core support barrel are dealt with in Chapter 4. The results of the examination and inspection are included in Chapter 5. Chapter 6 deals with the analysis of the failure mechanism and Chapter 7 with the core support barrel structural integrity, after the removal of the thermal shield. Chapter 8 summarizes the safety aspects of the thermal shield removal and finally Chapter 9 outlines inspection and monitoring.

2.0 Core Performance Evaluation

The Core Performance review concerns itself only with Chapter 3, i.e., the increased fast neutron fluence to the pressure vessel and the increased energy deposition rate due to neutron and gamma ray interaction.

The staff review regarding the removal of the thermal shield (in the Chapter 3 review by the Core Performance Branch) centered about (a) the increased fast neutron flux to the pressure vessel and the associated change in the RT_{NDT} versus the PTS screening criteria and (b) the increased neutron and gamma ray energy deposition (increased heat source) in the pressure vessel which results in a greater temperature gradient and a higher mean operating temperature versus its effect on PTS transients. In the following we shall examine each of these issues.

2.1 Increased Fast Neutron Flux to the Pressure Vessel

The DOT-4.3, an S_n finite difference neutron transport code, was used to calculate the azimuthal and the axial flux profile to the pressure vessel without a thermal shield. A long term axial peak of 1.14 was used with an

S_8 approximation for the S_n quadrature and a P_3 for the scattering cross section. The CASK cross section set was used which is known to yield conservative flux estimates when used with DOT-4.3. Lifetime average pin power distributions for the azimuthal model were calculated using PDQ-7. Similarly the axial model sources were based on an approximation of the expected time averaged axial power distribution. In this manner, the calculated peak fast neutron ($E > 1.0$ MeV) flux to the pressure vessel increased by a factor of 1.73. However, an additional perturbation was introduced with the drilling of crack arrest holes in the core support barrel. The flux perturbation cannot be calculated with the above scheme due to flux non-separability in the area of the holes. Hence, the effect of the holes was computed with a separate azimuthal calculation and the result was superimposed on the overall value of the flux. This is a conservative estimate. For a 6.6 inch diameter hole, the additional peak flux was higher by a factor of 1.24 over the uniform core support barrel flux. In reality the crack arrest holes are only 1.125 inches in diameter (four holes), hence, the above coefficient is conservative. The peak weld fluence of 5.0×10^{19} n/cm² occurs at weld 9-203, at 32 effective full power years. This corresponds to an RT_{NDT} value of 205°F, compared to a 300°F screening criterion for peripheral welds. The RT_{NDT} value is even lower for the lead plate, i.e., 197°F for plate C-505-2. The RT_{NDT} value of 300°F as the screening criterion for peripheral welds has been established by the staff and proposed to the NRC in SECY-82-465 on December 9, 1982.

2.2 Increased Neutron and Gamma Ray Energy Deposition

The removal of the thermal shield increases the neutron and gamma flux to the pressure vessel and, hence, leads to an increased energy deposition rate. This would result in a higher temperature gradient and a higher operating temperature. As indicated in SECY-82-465, the PTS is temperature sensitive and temperature difference dominated. On the other hand the higher operating temperature is beneficial to the vessel material properties. The maximum increase in the local vessel temperature was found to be 22°F at the outer surface. Increases of 17.5°F and 1.4°F were found at T/4 and the inner surface, respectively. The calculational method was similar to that used for neutron transport, i.e., was based on the DOT code.

The analysis of the effect of the increased energy deposition was based on the OCA-I code transient analysis with temperature distributions corresponding to levels with and without the thermal shield. The fluence values were arbitrarily set to a level which would cause crack initiation. This analysis indicated that the vessel without the thermal shield would have crack initiation later than the equivalent case with the thermal shield. This means that the beneficial effect from the higher temperature operation in the pressure vessel more than offsets the detrimental effect of the increased temperature gradient.

2.3 Core Performance Evaluation Conclusions

We have found that the estimated fast neutron fluence at the end of life and the increased energy deposition rate to the pressure vessel have been estimated with benchmarked and tested and therefore acceptable methods. With respect to the pressurized thermal shock, neither the value of the estimated RT_{NDT} nor the thermal gradient resulting from removal of the thermal shield will adversely affect the safe operation of the plant. We conclude that the proposed removal of the thermal shield at Millstone is acceptable from the viewpoint of increased neutron flux to the pressure vessel and increased energy deposition.

3.0 Materials Engineering Evaluation

The damage to the thermal shield support system at Millstone 2 consisted of cracking of the thermal shield to the extent that some pieces had broken off, the loss of some positioning pins and positioning pin lockbars. Associated with the thermal shield damage was cracking of the core support barrel in the vicinity of the thermal shield support lugs.

An indepth investigation of the failure mechanism has been pursued. It included investigation of hydraulic loads, structural response, metallurgical evaluation, design, fabrication, installation, and inspection data. The metallurgical results show that the cause of failure was high stress-low cycle fatigue, with no evidence of defective material or stress corrosion.

3.1 Cause of Failure/Inspection

The licensee has stated that, from the balance of the investigation, it appears certain that the damage was caused by large amplitude self-excited vibration. It is very likely that the vibration was made possible by deterioration of the thermal shield support system. The deterioration was probably preceded by loss of preload on positioning pins. The reasons for the loss of preload have not been specifically identified, but several factors have been examined and found to be capable of contributing. It is believed that a combination of the detrimental factors is the most reasonable explanation of the loss.

There are many details regarding the cause of failure of the thermal shield, thermal shield support system, and core support barrel that have not been explained fully. Nevertheless, the failure sequence described by the licensee is reasonable, assuming that preload on the thermal shield positioning pins was lost.

Three inspection methods were used to evaluate damage. These were: (1) visual (underwater TV), (2) Eddy current testing, and (3) Ultrasonic testing.

The loss of preload has not been satisfactorily explained beyond the statement that a combination of (unidentified) detrimental factors were the cause. Nevertheless, the staff agrees with the licensee that the core support barrel damage was caused by the thermal shield support system degradation and the resultant loads in the thermal shield/core support barrel system. Removing the thermal shield from the reactor internals will relieve a major source of loading on the core support barrel.

Because of the uncertainty that the visual inspection located all of the cracks in the core support barrel and to verify that additional degradation has not occurred, we recommend that an inspection of the core support barrel be performed at the next refueling outage. Furthermore, the sensitivity and resolution of the inspection system should be improved.

The staff notes that Figure 7.3-1 is either incorrect or confusing, as it shows two views that do not agree. It should be revised to clarify this situation. In addition, a minimum dimension should be identified for the remaining ligament.

3.2 Core Barrel Repair

The core support barrel was returned to service following repair of damage in the area of two of the thermal shield support lugs. The damaged areas have been inspected using nondestructive examination techniques, and repair methods utilized that were formulated to insure core support barrel integrity. An analysis of the repaired barrel has been completed, which shows that the original design criteria are met; stress levels remain within those allowed by the American Society of Mechanical Engineers (ASME) Code, Section III, 1977 Edition plus Addenda through Summer 1978.

3.3 Effect of Removal of Thermal Shield

The effect of the removal of the thermal shield on Pressurized Thermal Shock was of concern because it was felt that the increased fluence and gamma heating would be detrimental. It should be noted, however, that certain of the C-E plants are of very similar configuration (i.e., without a thermal shield).

Chapter 3, "Pressurized Thermal Shock" was reviewed first for the calculation of end-of-life RT_{NDT}, taking into account the added neutron fluence resulting from removal of the thermal shield. The report indicated that plate C-505-2 was limiting from PTS considerations. NNECO calculated an EOL RT_{NDT} of 197°F, well below the NRC PTS screening criterion of 270°F for base metal and axial welds. We checked the value of 197°F when calculated by PTS rules, i.e., the calculational procedure and the fluence, copper and nickel values are satisfactory. Of the other plates and welds, the beltline girth weld, 9-203, is nearly as limiting. Its EOL RT_{NDT} is 205°F when calculated by the PTS rule, but the screening criterion is 300°F for circumferential welds. The 205°F value was calculated using a copper content of 0.30%. In the December 9, 1977 report to the NRC giving the chemical composition for the beltline materials, weld 9-203 was reported to be made from two weld wire heat numbers. The copper content measured from the weld procedure qualification (PQ) samples were 0.37% for the wire used on the vessel ID and 0.23% for the wire used on the vessel OD. The surveillance weld was made to match girth weld 9-203, and the respective copper contents were reported to be 0.30% and 0.21%. Nickel content was 0.06% for the surveillance weld (both wires) but was not measured for the weld PQ samples. The EOL RT_{NDT} would be 258°F, calculated by the PTS rule, if copper was assumed to be 0.37%. Thus, for the purpose of checking EOL RT_{NDT} against the screening criterion, the choice of copper content is not critical.

The staff accepts the NNECO analysis of the effects of removal of the thermal shield on the end-of-life RT_{NDT} and agrees that Millstone 2 will meet the present PTS screening criteria throughout the presently-planned lifetime.

For the purpose of recalculating the pressure-temperature limits at the expiration of the present operating period, the copper content of weld 9-203 must be established. Preliminary information supports the use of 0.30% Cu, as proposed by NNECO.

Chapter 3 was also reviewed with regard to the effects of removal of the thermal shield on the temperature distribution in the wall that results from gamma heating. The NNECO submittal reports that the temperature deep in the wall will be 22°F higher and the temperature gradient will be about 26°F instead of 5°F after removal of the thermal shield. Because these two factors have opposite effects on the tendency to suffer crack initiation in thermal shock, NNECO made conventional PTS calculations of the critical crack depth as a function of time in the transient for two typical rapid cooldowns. The staff concludes that the results appear to be reasonable and agrees with the NNECO conclusion that the effects of thermal shield removal are slightly beneficial as far as thermal shock effects are concerned.

3.4 Material Engineering Evaluation Conclusions

1. The effect of removal of the thermal shield on the susceptibility of the vessel to pressurized thermal shock is negligible.

2. The staff agrees that the modification made will permit safe operation of the plant.

3. The staff recommends that the core barrel be inspected during the next outage. The inspection methods should be upgraded by using higher resolution television equipment and/or computer enhancement.

4. A revised Figure 7.3-1 in Millstone Unit No. 2 Thermal Shield Recovery Program Report should be submitted to clarify the record.

4.0 Mechanical Engineering Evaluation

The Mechanical Engineering Branch has reviewed the information in References 3 and 4 regarding the effects of the thermal shield removal on the structural integrity of reactor internals. Our evaluation concludes that the information presented by the licensee is adequate and acceptable to ensure that the stated design modification of the reactor internals will not compromise its original design margins, either to withstand flow-induced vibratory loads under various operating transients, or to resist postulated accident load such as under LOCA and SSE events. Our acceptance is based on the following:

1. The thermal shield removal will result in an increase in the reactor coolant system flow of less than 1%. However, in light of the 3.3% flow rate decrease currently experienced in Millstone 2 due to steam generator tube plugging, we conclude that the hydraulic loads on the reactor internals after the thermal shield removal will be within the limits that the internals had experienced during the preoperational and startup vibration tests.

2. Preoperational vibration monitoring and subsequent commercial operation of a plant, similar in reactor internals geometry and design to Millstone 2 except without a thermal shield, has shown a uniform core inlet flow distribution and no problems reported in flow-induced vibrations. Thus we concur that removal of the thermal shield will have negligible impact on internals flow distribution, and flow-induced vibration is unlikely to become a problem.

3. The licensee has conducted analyses regarding effects of normal operating loads, site specific seismic loads and asymmetric LOCA loads on the reactor internals with the thermal shield removed. The analytical results meet the ASME Code allowables used for the original design. Our evaluation concludes that this is justifiable for ensuring that the original design margin has not been reduced.

4. The repair of the Millstone 2 core support barrel consists of drilling four 1.125 inch diameter crack arrestor holes at the tips of the two through wall cracks. A core shroud/core barrel mock up test program is being conducted to determine the worst expected jet impingement magnitude and profile which could result from these holes, and an assessment of potential damage on the fuel is to be made. The preliminary results of tests and analyses indicate that the limits set by the original fuel vendor criteria are retained. We concur that such tests and analyses are necessary to ensure that no unacceptable safety concerns result from adding these holes in the core barrel of Millstone 2.

5.0 Summary

Since the licensee has restarted the plant under the provisions of 50.59, no specific licensing action is required. Our reviews indicated no unsatisfactory conditions or unresolved problems. However, the staff reccommends that the core barrel be inspected during the next outage.

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REFERENCES

1. Memo from P. H. Leech dated August 17, 1983 on Meeting - Millstone 2 Thermal Shield.

2. Memo from K. L. Heitner dated November 17, 1983 on Meeting - Thermal Shield Recovery Program.

3. Letter, W. G. Counsil of Northeast Utilities to J. R. Miller of NRC, with attachment report entitled, "Millstone Nuclear Power Station, Unit No. 2, Thermal Shield Damage Recovery Program, Final Report", B10968, December 12, 1983.

4. Letter, W. G. Counsil to J. R. Miller, B10981, December 22, 1983.