NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK



Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

June 6, 1973

Mr. Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Skovholt:

Re: Provisional Operating License: DPR-17
Docket No.: 50-220

On April 18-20, 1973 during the Spring 1973 refueling outage at Nine Mile Point Nuclear Station, Unit #1, the main steam isolation valves were leak tested, as required by the Station Technical Specifications. Once the results of these tests were evaluated, Division 1, Compliance was notified that three of the four valves failed to meet the leakage criteria of 12.9 SCFH.

The main steam flow from the reactor to the turbine is through two independent 24" lines. Each line contains two isolation valves, one inside the containment and one immediately outside the containment. The three valves that failed to achieve less than the maximum leakage of 12.95 SCFH were both outside isolation valves and one inside isolation valve.

Under containment design basis accident condition therefore one main steam line would remain essentially leak tight while the other line would leak at a rate of 25.5 SCFH into a closed system eventually terminating at the stack.

The MSIV's are intended to be closed during the design basis loss-of-coolant accident with a low leakage rate to assure than any significant release of fission products is retained within the containment system.

The maximum allowable test leak rate from the containment is 1.5%/day as a pressure of 35 psig. This was derived from the maximum allowable accident leak rate of about 1.9%/day when corrected for the effects of containment environment under accident and test conditions.





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Although the dose calculations suggest that the allowable test leakage rate could be increased to about 3.0%/day before the 10 CFR 100 guideline thyroid dose limit would be exceeded. The limit was established at 1.5%/day to provide an adequate margin of safety to assure the health and safety of the general public. In addition the operational limit was established as the multiple of the allowable test leak rate and .75, thereby providing a 25% margin. Therefore even though the test leakage was higher than the operational limit the value is mitigated by the safety margins and conservation used to derive the operational limit, and it can be concluded that even with the slightly higher leakage rates thru the one line, no undue hazard would have been presented to the general public in the event of a containment design basis accident during the previous operating cycle.

The accident analysis for a main steam line break outside the drywell concerns itself with two primary considerations:

1. The coolant inventory of the Reactor Vessel,

2. The radiological releases to the environment.

Both cases depend upon the closure time of the main steam isolation valves. In order to maintain coolant coverage of the reactor core a maximum closure time of 10 seconds is assumed in the analysis. Using this closure time for one valve in each main steam line the reactor core remains covered. The radiological releases to the environment are within limits provided a maximum closure time of 11 seconds is maintained. However, this is also based upon the primary coolant radioactivity concentration limit of 25 μCi total iodine per gram of water.

The actual closure times for the inside isolation valves are below the analysis maximum values used. In addition, the yearly average primary coolant radioactivity concentration was .25 μCi total iodine per gram of water. In conclusion, no undue hazard was presented to the general public if a main steam line break had occurred during the operating cycle.

Results for the four valves are as follows:

| Valve Numbers | Leakage SCFH Before | Leakage SCFH After |
|---|---|---|
| #11 inside | 25.5 | 9.90 |
| #11 outside | 1116.1 | 9.55 |
| #12 inside | 5.0 | 5.00 |
| #12 outside | 64.4 | 3.75 |

Following core refueling, repairs were effected to those leakage valves using machine lapping techniques. A blueing of the seats was made before and after repairs. The valves were then retested to determine their leakage. All valves meet the 12.9 SCFH prior to restart of the reactor.

Very truly yours,

P. Allister Burt

P. Allister Burt
General Superintendent
Nuclear Generation

PAB/cm