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ATOMIC ENERGY COMMISSION
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Files (Docket No. 50-263)
THRU: D. L. Ziemann, ORB #2, L
D. R. Ziemann

MONTICELLO - TECHNICAL SPECIFICATION CHANGE NO. 5

By letter dated February 2, 1973, Northern States Power Company (NSP) requested changes to the Technical Specifications involving:

1. Rod Block Monitor

2. Core and Containment Cooling System

3. Reactor Recirculation System Cross-tie Valve Interlocks

4. Refueling Interlocks - Extended Core and Control Rod Drive Maintenance

Additional information relative to the requested changes was submitted by NSP in a letter dated February 20, 1973. Following is a discussion of each of the proposed changes.

1. Rod Block Monitor (RBM)

According to the basis provided on page 67 of the Technical Specifications, withdrawal of the control rod with the highest reactivity worth with no rod block action and reactor power level below 70% results in MCHFRs greater than 1.

The RBM provides local protection of the core; i.e., it prevents critical heat flux in a local region of the core for a single control rod withdrawal error from a limiting control rod pattern(2).

The proposed technical specification change to allow the upscale rod blocks to be bypassed whenever the reactor is operated below 30% of rated power is necessary, according to NSP representatives, to prevent false rod block signals from spurious noise signals at the low power and flow conditions where accurate nuclear instrumentation calibration is impractical.

In a typical startup situation, the flow rate is generally in the vicinity of 20% and as required by the Technical Specifications, Table 3.2.3, item 4, the corresponding flow dependent upscale trip signal is ≈55% of rated power level.

NSP representatives stated, however, that continuous fast withdrawal of the most reactive control rod with initial power level less than 30% would result in a local transient peak power less than the 55% upscale trip point; and, therefore, the rod block monitor upscale signal would not normally block the rod withdrawal. Although we cannot confirm this behavior quantitatively, the description is reasonably similar to a worst case analysis for a control rod withdrawal while at 100% of rated power and flow as presented in a General Electric topical report[3] where the local power perturbation in the vicinity of the withdrawn control rod is compared with average core power. We have concluded that the rod block monitor upscale - downscale signals are not required to protect the core at power levels below 70% and that if the signals are bypassed as proposed below 30% rated power level to prevent spurious RBM trip signals at the low power levels, the difference between 30 and 70% would represent an acceptable operating margin to a reactor technical limit; i.e., MCHFR < 1.0. Therefore, the Technical Specifications may be changed as proposed.

2. Core and Containment Cooling Systems

To reduce shutdown time, NSP has proposed technical specifications to permit draining the torus for inspection and maintenance while simultaneously performing control rod drive maintenance. We have reviewed similar proposals[1] for Dresden 2 and 3 and concluded that one control rod drive housing or instrument thimble can be opened at any one time while the suppression chamber is drained because there is sufficient water in the storage pool above the core to prevent uncovering the core if maximum leakage occurs through an open control rod drive housing. To further minimize the possibility of draining water from the core during removal of control rod or instrument thimble, maintenance procedures have been established[4] to minimize the time water could drain from the core while performing the maintenance activity. On the basis of the above, we have concluded that the Technical Specifications may be changed as proposed.

3. Reactor Recirculation System Cross-tie Valve Interlocks

The technical specification changes to require that the recirculation system cross-tie valve interlocks be operable with bo h recirculation pumps in service and to permit reactor operation with inoperable recirculation system cross-tie valve pump interlocks when both recirculation pumps are in service, if at least

one of the two cross-tie valves is maintained fully closed, have been proposed by NSP to limit the double-ended recirculation line break to the equivalent of 4.2 square feet. The change assures that the equivalent size of a double-ended coolant recirculation pipe break is no greater than the limiting break, 4.2 square feet, which satisfied the AEC Interim Acceptance Criteria (2300°F peak clad temperature) for Emergency Core Cooling Systems as described in the NSP report dated September 21, 1971. The pump recirculation system cross-tie valve interlocks will cause automatic closure of the cross-tie valves during normal operation whenever both coolant recirculation pumps are in operation. However, if the interlocks are inoperable, administrative procedures will be substituted to assure that at least one of the two cross-tie valves is fully closed. We note that (1) the proposed changes apply only to two loop reactor operation with both recirculation pumps in service, (2) the proposed changes are consistent with the emergency core coolant injection requirement to assure emergency coolant access to the core through the undamaged coolant loop, and (3) NSP has verified that automatic closure of the cross-tie valves following coolant pipe rupture with only one recirculation pump is unaffected by the proposed changes. We have concluded that the technical specification changes, as proposed by the applicant, should be made (1) to prevent core damage (i.e., peak fuel clad temperatures greater than 2300°F) by limiting the equivalent pipe break size in the event of the very unlikely double-ended rupture of the coolant recirculation pipe while at power levels in excess of 70% and (2) to permit emergency core cooling through the undamaged loop.

4. <u>Refueling Interlocks - Extended Core and Control Rod Drive Maintenance</u>

During core alterations the reactor mode switch must be locked in the "Refuel" position according to Technical Specification 3.10.A. NSP has proposed to change this requirement by adding a new specification (3.10.E) so that control rod drives with or without control rods can be withdrawn by bypassing the refueling interlock input signal from the control rod when the fuel assemblies in the cell containing (controlled by) that control rod are removed from the reactor core[5]. With 12 spare control rod drives, there is normally no need to have more than one drive removed from the reactor vessel control rod drive housing at any one time. The usual procedure is to remove a control rod drive and replace it with a spare on a one-for-one basis. Since fuel assemblies require the inserted control rod for guidance into the correct position in the core, it is an administrative prerequisite for fuel loading that a blade guide be positioned in the cell to guide the control rod as it is driven into the core. In other words, the control rod drive must be restored to the operational condition and the

control rod drive interlocks must be restored to satisfy requirements for loading fuel into a selected core cell to meet the new technical specification requirement that allows the interlock to be bypassed only when the adjacent fuel is removed.

Accidental dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod will not cause the neutron population to reach the self-sustaining level as noted in the technical specification bases for the refueling interlocks.

With the understanding that the core becomes less reactive as each cell is emptied (fuel removed) and that the reactivity worth of a control rod in adjacent fueled cells does not exceed the maximum control rod worth for the fully loaded core as calculated and reported(5) by NSP during a February 23, 1973 telecon, we have concluded that there is no technical basis to restrict the number of control rod drives that may be removed and blanked off at the housing. We note also that without the surrounding fuel bundles or the temporary blade guide (tuning fork) in place, the control rods serve no safety function since they could not be driven into the core without risk of damage to the fuel and poison control rod.

We have concluded that the proposed change to Technical Specification 3.10.E-1 may be made as modified.

On the basis of our review, as described above, we have concluded that the proposed changes do not present significant hazards considerations not described or implicit in the Monticello Safety Analysis Report and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, the Technical Specifications should be changed in the manner described.

James J. Shea

James J. Shea
Operating Reactors Branch #2
Directorate of Licensing

Enclosure:
References

cc: See next page

cc w/enclosure:
Northern States Power Company
AEC PDR
Local PDR
RTedesco, L:CS (2)
DJSkovholt, L:OR
TJCarter, L:OR
RO (3)
DLZIemann, L:ORB #2
JJShea, L:ORB #2
RMDiggs, L:ORB #2
MJinks, DRA (2)

REF NCES

1. AEC approval of Changes No. 17 - License No. DPR-19, Dresden 2, and No. 9 - License No. DPR-25, Dresden 3, dated March 17, 1972, "Control Rod Drive Maintenance with Drained Torus".

2. Rod Block Monitor - FSAR page 7.4-11.

3. An Analysis of Functional Common-Mode Failures in GE BWR Protection and Control Instrumentation - page 97, NEDO-10189 70 NED 16, July 1970.

4. Supplementary Information Supporting Specification Change Request dated January 31, 1973 - NSP letter dated February 20, 1973.

5. NSP-AEC telecon on February 23, 1973.

   "With a voided cell (4 bundles surrounding a control rod removed from core) and the control rod of the highest reactivity worth withdrawn, K_{eff} for the core decreases by about 1.5% according to recently completed NSP calculations."