MAR 5 1969

R. S. Boyd, Assistant Director for Reactor Projects, DRL
THRU: S. Levine, Assistant Director for Reactor Technology, DRL /S/

NINE MILE POINT NUCLEAR STATION - DOCKET NO. 50-220

The reactor internals portion of our contribution to the ACRS report for the subject facility is enclosed.

/S/
A. W. Dromerick, Chief
Containment & Component Technology Branch
Division of Reactor Licensing

RT-203A
DRL:SSTB:JPK

Enclsoure:
Reactor Internals Report

cc w/encl.: R. Tedesco
V. Stello

bcc: S. Levine
R. C. DeYoung
A. W. Dromerick
J. P. Knight

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Reactor Internals

The applicant has confirmed that the integrity analysis for the Nine Mile Point reactor vessel internals is identical to that which was given in Amendment 12 to the FSAR for the Oyster Creek Nuclear Power Plant Unit No. 1 (response to question 6 - Second Supplement to the FSAR). Although some results are slightly different for Nine Mile Point because of changes in steam line orificing and design steam flow, there are no significant differences in loadings.

The Nine Mile Point reactor internals are designed to withstand the loads due to normal operating transients and steady-state conditions combined with the design basis earthquake loading. The applicant states that the primary and secondary stress requirements of Section III, ASME Code, for Class "A" vessels are met for these combined loading cases. We find these limits acceptable.

Under accident conditions, which include a recirculation line break or a steam line break but no simultaneous seismic loads, the deflections and deformations were limited to values at which the reactor internals maintain "functional capability". This term is defined by the applicant as the maintenance of a "cage-like configuration" to the extent that design

control-rod-drive scram capability and design core-spray functions are not affected. The seismic loadings on the core structure were based on the dynamic analyses of the reactor vessel only. As the basis for this approach the applicant cites a dynamic analysis performed for a similar plant in Japan (TSURUGA) which included the reactor internals and which indicated that the response of the core structure was 20 to 40 percent lower than that of the vessel. While we do not feel, as the applicant does, that this is a conservative approach, we have examined the magnitude of seismic loadings with respect to the conservatisms used in modeling the accident conditions and the conservative stress limit criteria used in this application. On the basis of this evaluation we find that adequate margins of safety exist for these combined accident and seismic loading conditions. Our seismic design consultants agree with this finding

The applicant states that the design of the reactor internals was checked analytically for vibration and hydro-dynamic stability. Recognizing, however, the difficulty of accurately modeling the internals system the applicant further states that they are relying on the vibration measurements being done on the Oyster Creek No. 1 Nuclear Plant to bear out the applicability of their analytical efforts. While we generally concur with this approach we take note of this tie to the Oyster Creek test schedule and feel that the applicant should make a firm commitment to do likewise. We also feel that the applicant should make a firm commitment to take any corrective steps shown necessary by the Oyster Creek vibration tests before completing the test program for the Nine Mile Point facility.