DEC 1 8 1968

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

INDIAN POINT -- UNIT NO. 3 (DOCKET NO. 50-286)

Enclosed is a report prepared by S. Pawlicki on his assigned areas of review for the Indian Point No. 3 facility.

A draft copy of the enclosure was given to J. Murphy of Reactor Projects No. 1, on December 12, 1968, for his guidance in preparation of the ACRS report.

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

RT-15A
DRL:C&CTB:SSP

Enclosure:
As stated above

cc: D. R. Muller, DRL
J. Murphy, DRL (3)

bcc: S. Levine, DRL
R. C. DeYoung, DRL
A. W. Dromerick, DRL
S. S. Pawlicki, DRL

Distribution:
Suppl
DRL Reading
C&CTB Reading
ADRT Reading

8111200338 681218
PDR ADOCK 05000286
P PDR

OFFICE ▶	DRL:C&CTB	DRL:DADRT	DRL:ADRT			
SURNAME ▶	SSPawlicki AWDromerick	RCDeYoung	SLevine			
DATE ▶	12-17-68	12-17-68	12-18-68			

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE : 1966—O-214-629

INDIAN POINT - UNIT NO. 3

DOCKET NO. 50-286

Reactor Internals

The reactor internals will be designed to meet normal design loads of mechanical, hydraulic, and thermal origin plus operational basis earthquake loads (OBE).* Stress limit criteria used are as established in Section III of the ASME Boiler and Pressure Vessel Code.

The reactor internals will also be designed to withstand the concurrent blowdown and design basis earthquake (DBE)** loads, as indicated by the applicant in Supplement 1, Section 15. Primary tensile stresses under such load combination will not exceed stresses corresponding to 20% of the uniform strain at temperature, while the allowable deflection limits will be about 50% of the loss-of-function deflections for the specific components. We consider these stress and deformation limits to provide adequate margins of safety, since they are basically the same as the criteria recently accepted for other PWR plants.

Reactor Coolant System

Section III of the ASME Pressure Vessel Code will be used to design the reactor vessel, pressurizer, coolant pump casings, and the steam generator.

---

* OBE - The smaller earthquake, about one-half the design basis earthquake (DBE). The plant will be designed to continue normal operation during an operational basis earthquake.

** DBE - The larger earthquake. The Class I (seismic) items must retain their functional capability during a DBE.

The reactor coolant piping design will be analyzed in accordance with the requirements of USA S.I. B31.1 Code for Pressure Piping. The applicant stated that the analysis will take into account the interrelation of the primary system components, piping, and supports.

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification will be prepared by the manufacturer to assure compliance with the stress limits of Section III for the reactor vessel, steam generator, pressurizer, and pump casing. Westinghouse independently will review these stress analyses. A similar analysis of the piping will be prepared by Westinghouse or for Westinghouse by a qualified piping analysis contractor.

The reactor coolant system, and all other Class I (seismic) mechanical systems, will be designed to withstand normal design loads of mechanical, hydraulic, and thermal origin plus operational basis earthquake loads (OBE) within normal code allowable stresses. In addition, as stated in Amendment 1, Class I systems and components will be designed to withstand the concurrent blowdown and design basis earthquake (DBE) loads. Primary membrane stresses under such load combination will not exceed stresses corresponding to 20% of the uniform stress at temperature.

We conclude, on the basis of our evaluation, that the design criteria proposed for the reactor coolant system provide adequate margins of safety.

Reactor Vessel Thermal Shock

Our general review of the thermal shock problem is continuing. We are still uncertain that presently available experimental data, and the analytical techniques of elastic fracture mechanics, can clearly demonstrate that the reactor vessel will maintain its integrity under thermal shock conditions.

The uncertainties in the analysis of the thermal shock effects on the reactor vessel are of three origins:

a. The heat transfer calculations leading to temperature and stress distributions through the vessel wall, as a function of time,

b. The experimental data on fracture toughness, and

c. The analytical techniques of elastic fracture mechanics.

There is general agreement among the reactor manufacturers, based on results of fracture mechanics analyses, that an initial small crack, which could be present at the vessel beltline, would propagate under the thermal shock stress conditions. As of October 1968 the extent of crack propagation, assuming an initial circumferential crack and cooling water temperature of about 70°F, has been calculated as follows:

B&W	55% penetration at	600 sec
CE	40% penetration at	1000 sec
W	60-80% penetration at	1000 sec

We have received recently an additional written submittal from Westinghouse, based on new fracture toughness data. The report concludes that considering the conservative lower fracture toughness band, any crack propagation is expected to be less than 32 percent; and, therefore, the integrity of the reactor vessel will be maintained throughout the life of the plant. We are presently reviewing that report.

At the present time, it appears that there should be no danger of vessel failure until several years of vessel irradiation. The Heavy Section Steel Technology Program at Oak Ridge National Laboratory, due for completion by 1973, will provide additional data on material properties. Westinghouse is also participating in Euratom-funded fracture mechanics program to obtain irradiated fracture toughness properties. Furthermore, even if it should be shown that the vessel might crack, there appear to be suitable engineering solutions that could be employed if needed.

We have also reviewed the effect of potential fracture of the vessel due to flooding of reactor cavity, if the Post Loss-of-Coolant Accident Protection System (PLOCAP) is added, as described in Section 4.2.2 of this report. Since the proposed flooding system would be actuated by the combination of safety injection and accumulator low pressure signals, flooding of the cavity would occur only after subjecting the interior of the vessel to the thermal shock stresses, and at low reactor pressure. We concluded, therefore, that flooding of the reactor cavity would not significantly add to the reactor vessel fracture hazards.