U.S. Nuclear Regulatory Commission

In the Matter of
CPCo. Midland Plant
Units 1 & 2

Docket Nos.
50-329 OL
50-330 OL

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

OPERATORS LICENSE CONTENTIONS OF INTERVENOR STAMIRIS

6/18/82

I am an accepted intervenor in the O.L. proceeding regarding soils related matters. On March 28, 1982 I sought to expand my O.L. participation to other matters for the good cause mentioned therein. In subsequent conference calls of OM-OL parties, I was directed to submit contentions for O.L. consideration by June 21, 1982.

I contend that:

1. Bechtel's design audit (discussed by J. Cook at 5/20/82 ACRS meeting) constitutes self-monitoring which cannot provide reasonable assurance of safety and conservatism in the plants implemented design, and should not be relied upon for such assurance.
2. The NRC has allowed significant departures from original design criteria on the basis of financial hardship to the applicant, as seen in:
   a. Soils remedial measures
   b. Reactor pressure vessel remedial support system
   c. Reactor vessel welds

These departures do not provide adequate margins of safety for plant operation

and place undue consideration on financial hardship as opposed to safety considerations.

3. The NRC's economic analysis presented in the February 1982 DES misrepresents facts to the public in an attempt to justify Consumer's financial investment because it:

   a. fails to consider the $3.39 billion construction costs although these costs will be borne by ratepayers

   b. estimates only $235 million for decommissioning when Consumer's estimated about $500 million to decommission Big Rock and Palisades in 1980

   c. estimates about a 36 year lifespan, despite the shorter life expectancy of unit one due to the defective weld (p. C-10 SER)*

   d. uses Consumer's estimated annual growth rate of 3.2% while recent studies by Michigan's Attorney General project a 1-1½% longterm growth rate in Michigan

   And such a faulty analysis cannot be allowed to stand unchallenged by a rule which prevents one party (ie intervenors) from considering financial issues, when another party (the NRC) has already raised those issues.

4. The NRC has not enforced its own 10 CFR 50 Appendix B regulations adequately, and this pattern of NRC leniance has aggravated the Applicants Quality Assurance problems over the years, resulting in an unsafe nuclear plant.

5. The NRC practice of accepting significant design changes to the PSAR/FSAR, after the construction in question has been completed, has allowed much of the Midland plant to be built without pre-set design requirements, resulting in a plant considerably less safe than the approved construction permit design.

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*all page nos. refer to 5/82 SER, but do not exclude other possible uncited references.

6. The Applicant's frequent failures to follow design requirements, as reflected in 55e construction deficiency reports and I&E inspection reports, cannot provide assurance that inaccessible or unverified plant areas have been built as designed to meet safety standards.

7. Utility/NRC "proceed at your own risk" agreements as practiced at Midland are contrary to Atomic Energy Act mandates, because financial "sunk costs" are considered in the end by the NRC, rather than basing ultimate decisions on safety alone as implied. These end results negate the element of financial risk to the utility, and instead risk public health and safety by compromising original safety standards. Such weighing of financial investments against safety standards necessitates value judgements by the NRC that go beyond the authority vested in them to protect public health and safety only.

8. The NRC/CPC reporting system intended to allow plant workers to raise concerns or criticisms about inadequate workmanship or practices is ineffective, and therefore plant safety issues go unresolved.

9. It is improper for Bechtel workers to have to sign a promise not to divulge information about the nuclear plant to the public as an initial job condition, because it tends to intimidate workers from raising safety concerns when frustrated with internal reporting system limitations or repurcussions.

10. The NRC has placed undue consideration upon Consumer's financial and scheduling needs resulting in significant reductions to public health and safety standards which were wholly preventable.

11. Consumer's installation of the Unit I reactor subsequent to the identification of its seriously defective condition (weld WF-70), the failure to follow proper weld testing procedures, and the failure to include this "most limiting" WF-70 material in surveillance measurements for reactor welds, represents a pattern of careless disregard of safety principles which cannot assure safety or trustworthiness and should not be accepted by the NRC (p. 5-17 to 5-24).

12. The NRC practice of allowing Consumer's exemptions from minimum safety requirements becauee of economic hardship cannot be accepted on the basis of Consumer's commitment to future monitoring (ISI programs), because Consumer's has not proved themselves worthy of this trust. Such exemptions include

    a. reactor vessel weld conditions (p. 5-14 5-25)

    b. relief from code testing requirements on safety pumps and valves (p. 3-35)

    c. postulated high energy pipe break criteria (p. 3-13)

    and these exemption/inspection arrangements jeopardize public safety.

13. Consumer's failed to apply required single failure criteria throughout the plant design and construction as seen in

    a. Small break/Reactor Coolant Pipe Interaction (79-03)

    b. Letdown Cooler Supports (30-01)

    c. HELBA restraint design (80-03)

    d. CCW deficiency (80-06)

    e. NSSS Seismic/LOCA analysis (80-07)

    f. Auxilliary Building Seismic Analysis (81-02)

    g. Shear reinforcement lacking in containment (81-05)

    h. Reactor Cavity Cooling deficiency (81-06)

    i. AFW Valve System (82-04) and Header design issues and related AFW (82-06)

    resulting in a nuclear plant unlikely to withstand design basis accident conditions.

14. The 1978 identification of extensive deficiencies in the procurement system for seismic and environmental qualification of equipment (78-10) should have prevented the following equipment qualification procurement errors;

    a. ITE Gould, Class 1E equipment (81-04), Gould starters (1979)

    b. Long electrical cables (81-07)

    c. Inadequate station batteries (1979)

d. Limitorque terminal strips (81-01)

e. MSIV Equipment (82-01)

f. AFW Power Supplies (82-03)

g. Wiring insulation deficiencies (82-02)

These equipment qualification deficiencies combine with design deficiencies of contention 13 to result in a plant unlikely to withstand design basis accident conditions.

15. The Bechtel errors in Seismic calculations, discovered as a result of reactor vessel support modification investigations, had extensive generic implications which should have precluded further plant construction involving siesonic parameters until the safety significance of these errors were resolved.

16. The Oct 23, 1981 CPC response to the 1978 identification of the accident condition unavailability of pressurizer heaters during plant hot shutdown does not provide the margin of safety required in original design documents.

17. The NRC states in the SER that "because the design of a majority of the Seismic Category I Structures was completed before 1973" their load combinations do not meet current NRC requirements (p. 3-21). Yet new design information regarding Seismic/LOCA design criteria was submitted by Bechtel in 1981 as reflected in the new design responsibility matrix. Therefore a waiver of current NRC Seismic requirements for concrete safety structures is not warranted for these reasons and does not provide adequate safety for design basis accidents.

18. The containment structures were designed for saturated plant fill soil conditions. The permanent dewatering system changes these conditions and the resultant soil-structural interaction effects, thereby reducing the ability of the containment structures to perform their intended safety function under design basis conditions.

19. The containment structures are subject to so many design and construction deficiencies that taken together they render the reactor containment system

incapable of performing its intended safety function. These deficiencies include

a. the effects of inadequate loading combinations (contention 17) and NSSS Seismic/LOCA deficiencies

b. the effects of dewatering (contention 18)

c. tendon sheath omissions (1977)

d. post-tensioning errors (1979)

e. RVP potential bolt failure stresses (further missle effects)

f. RVP support modification effects

g. lack of adequate shear reinforcement (81-05), cooling system deficiencies (79-07, 81-06)

h. failure to postulate containment penetration pipe break effects

i. 1974 Unit II fire effects (74-01):

1) bulge in containment liner

2) bulge in blade steel

3) protective coating damage

4) possible concrete damage

5) possible damage to horizontal rebar at floor

20. The extensive problems concerning RPV anchor bolts and LAQT high tension bolting in support of safety systems represent serious safety degradations could have been prevented by proper procurement qualifications and QA inspections of supplied materials.

21. The remedial reactor vessel support system intended to compensate for failed anchor bolts does not meet original safety standards and is unlikely to provide adequate protection against reactor internernal forces under design basis accident conditions.

22. The defective bolting in support of reactor coolant pumps results in a seriously degraded safety system, which taken in conjunction with other cooling system

weaknesses results in an unreliable safety system under design basis accident conditions.

23. Welding on class 1 and class 2 piping does not meet current safety standards (p. 5-12 to 5-14).

24. Deficient welding conditions and practices have not been adequately addressed by QA/QC supervisors upon the request of QC personnel, and unsafe welding conditions remain uncorrected.

25. The absence of design features for radiation exposure control during plant decommissioning and the absence of a specified decommissioning plan constitutes a public health and safety hazard.

26. Numerous design and construction deficiencies combine to make the decay heat removal system unreliable for performing its intended safety function. These deficiencies include:

    a. B&W system sensitivity issues

    b. AFW System deficiencies (82-03, 04, 06)* and piping not all to C. I standards

    c. Containment coolers water supplies (79-02)

    d. Reactor coolant pump defects (79-03, 80-09)

    e. CCW system deficiencies (80-06)

    f. Required manual valve operation for emergency boration and auxiliary pressure spray, and location of manual DHR valves (p. 5-32)

    g. Boration capabilities (79-11)

    h. Unusual corrosion in BWST stainless steel piping (1979) and generic implications

    i. Soil settlement effects on reliability of BWST, SWS, piping, and shared UHS

    j. Reactor Cavity Cooling System deficiencies (79-07, 81-06)

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* All year-numbers refer to 55e reports, not to exclude other possible references.

27. Despite the extensive deficiencies and reliability questions associated with the DGB onsite power supply due to soil settlement problems, the combined offsite/onsite blackout power failure accident is not a postulated design basis event for safe shutdown and this represents a serious unconsidered threat to public health and safety.

The AFW system and a turbine driven pump are not designed for and cannot be relied upon (see contention 26) to provide sufficient cooling water for this essential safety function from the non-category I condensate tanks (p. C-17).

28. An extensive pattern of electrical errors and deficiencies cannot provide assurance that these and other electrical systems will function properly to perform their essential safety function. These include:

a. NSSS component wiring (80-02)

b. ECCAS wiring (80-03)

c. MSIV actuators (82-01)

d. AFW valve power supplies (82-03)

e. improperly qualified electrical equipment (contention 14)

f. cable spreader room designed 50% too small

g. ongoing electrical problems (80-81 SALP)

29. Pressurized thermal shock probability, and over-pressurization as discussed in I.N. 82-17 create a hazard aggravated by deficiencies in other safety systems which has not been adequately addressed in an integrated fashion to assure that public safety will be protected in the event of an accident.

Respectfully Submitted,

Barbara Stamiris

Barbara Stamiris
5795 N. River Road
Freeland, MI 48623

cc: ASLB Judges
W. Paton, NRC
M. Miller, CPC
Secretary, NRC
T. Devine, GAP