UNITED STATES NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:)
)
HOUSTON LIGHTING AND POWER CO.,) Docket No. 50-466
(Allens Creek, Unit - 1)) May 25, 1979

JOHN F. DOHERTY'S ADDITIONAL CONTENTIONS AFTER ALAB-535

John F. Doherty, Intervenor in the above construction license proceedings submits the following contentions in response to the Board's Order of April 11, 1979. Intervenor was granted until May 25, 1979 to submit contentions by a succeeding Board Order. These contentions were not submitted previously because they either had nothing to do with change in plant design or were not based on entirely new evidence. Some arose due to events during the period granted by the Board to submit contentions.

AMENDED CONTENTION

In part #5 of the Board's Order of April 11, 1979, the Board required amendment to Intervenor's Contention #4, which read:

> "The ACNGS Applicant should be required to maintain flexibility such that design features required by the resolution of the ATWS generic issue can be incorporated in the design without modifications to main components of the nuclear steam supply system* during construction (1) to avoid additional costs, and (2) to assure full implementation of the generic resolution."

Intervenor amends the contention at the asterisk (*) by adding "and containment building structure" and by adding the below to the statement:

. Specifically, the design should be able to include additional space for the Standby Liquid Control System (SLCS), its test tank and pumps. In addition it should include space and a pathway for an enlarged delivery pipe for the SLCS to the reactor core with its rquirements for reactor vessel penetration. The containment building should be designed to support the weight that this enlarged system would entail, too. Intervenor interprets NUREG-0460 7.3 to mean the main ATWS mitigation will be enlargement of the SLCS system. 551 006

CONTENTIONS

As of May 23, 1979, there were eight (8) contentions of this Intervenor either accepted or in controversy. Hence numbering of contentions begins with "9".

9. Intervenor contends Applicant's safety systems contain many non-safety grade equiptment items. This issue was raised as a "Board Notification" by the Office of Nuclear Reactor Regulation. Applicant's Table 3.10-1 of the PSAR shows a large number of components which are qualified under 1EEE-323 and 1EEE-344 but not under 1EEE-279. Particularly significant to transient mitigation are level switches, pressure switches, pressure indicators, radiation monitors, trip units, electronic modules for control rod drive units, valve motor operators, temperature switches, indicators and elements, recirculation valves, and motors. Intervenor contends the use of these components of below safety grade violates General Design Criteria (GDC) #29, and 10 CFR 55a(h) and entails risk to his health and safety interests. Intervenor request that the Board enforce GDS #29 and 10 CFR 55a(h) on the equipment listed here and in Table 3.10-1, and equipment not listed that is used to mitigate transients.

10. Intervenor contends that his health and safety interests are not sufficiently protected because applicants diesel generator system to the High Pressure Core Spray (HPCS) and to the rest of the nuclear plant is unreliable in start up and operation. Specifically, Applicant's system is identical (largely) to one that according to NUREG-0660 (Feb. 1979) produced 122 start-up failures and which included 54 of these failures among BWR licensees from 1969 to 1979, and according to <u>Nuclear Safety</u> 19(1), pg. 81, had 74 "Reportable Occurences" in 1976 and 76 (<u>Nuclear Safety</u>, 20(1), pg. 84) in 1977. Further that diesel generators are subject to other than inherent failure through worker error (<u>Oyster Creek</u>, December 1975), poor administrative practice,

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i.e. fuel oil stored with lubricating oil and used as a lubricant (Brunswick, Unit-2, October 1975) and outside storage of diesel generator's prior to plant start-up, resulting in water logging (Ft. St. Vrain, June, 1975).

NUREG-0660 has concluded that interpretation by the utilities of the NRC regulatory guides on diesel generators has varied widely and been a problem source. Intervenor contends that the data above show the need for either (a) a third generator be installed for both the HPCS and a third for the balance of the plant systems, (b) a higher standard than 93 starts per 100 be used in requiring additional information to the NRC, (General Electric Technical Specification 4.8.1.1.4 for BWRs) and that (c) Surveillance Rquirement 4.8.1.1, should require an every 3 day surveillance, and all other surveillance times halved in view of the serious consequences of power failure to a nuclear plant.

11. Intervenor contends that danger of a spent fuel pool loss of coolant accident (LOCA) are not fully accounted for in Applicants plan to store large amounts of spent fuel rods in the spent fuel pools both in the containment building and the radwaste building. There is danger of spent fuel pool heat-up and evaporation or evaporation induced criticality event if all personnel must leave the ACNGS because of a serious accident at a nearby nuclear plant ("ACNGS Unit-2") or due to a more serious accident at South Texas Project Unit 1 and 2 or Blue Hills Unit 1 and 2. If all personnel must leave, the reactor can be shut-down, but the spent fuel pools would be dependant on permanent replenishment of water from coolant and reactivity control systems much less reliable than the reactor's systems. Evacuation of some persons as a result of radio-activity occured in March 1979, at the Three Mile Island incident, although these persons could return. Intervenor contends that his economic, safety and health interests will be injured if the spent fuel pool experiences a transient or other accident even if its is many years following his evacuation from the area, because contamination from more than six (6) years of spent fuel (ACNGS has room for ten (10) years) may contaminate areas and person previously spared. Applicant

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thus should be required to locate storage for spent fuel available as soon as the first fuel loading can be removed safely from the site.

12. Intervenor contends the Rod Pattern Control System in the Instrument and Controls systems of the proposed ACNGS is not reliable. The operators of Dresden Unit-3 (a G.E.,BWR) reported the system inoperable for 54% of start-ups in 1972. Quad Cities Units 1 and 2 were operable in but 74% of the start-ups, and Millstone Unit 1 reported this system failed in 172 of 245 start-ups in a 16 month period beginning in 1971. Further, 34.6% of "Reportable Occurences" in BWR reactors in 1977, were in the Instrumentation and Controls area (NUREG-0483, Page 4-7). The Average Power Range Monitor (APRM) used to detect surplus neutron flux in this system is not highly reliable. Power Range Insturments contributed to 36 "Reportable Occurences" in BWRs in 1977, and 17 in 1976, (Nuclear Safety, volumes 19(1) and 20(1), 1978 and 1979, pgs. 84 and 82 respectively). Most recently a rod block monitor was inoperative during start-up of the Brunswick-2 reactor (Sept. 4, 1978) due to a failed integrated circuit. Petitioners contend danger to their health and safety interest by a reactivity insertion accident during start-up unless Applicant installs a more reliable system than this one.

13. Intervenor contends Applicant's Containment Emergency Sump Pump will not function reliably because during a loss of coolant accident (LOCA) thermal shielding and insulation may be ripped off or otherwise released or separated from in-containment building piping, where it would block off the drain of water, preventing it from being recirculated for cooling by the sump pump. and this would degrade the effectiveness of the Emergency Core Cooling System (ECCS). This would endanger Intervenor's health and safety. This issue has been part of Task # C-3 in the Office of Nuclear Reactor Regulation as "Insulation Usage Within Containment". Since issues have been raised by Staff on Applicant's ultimate Heat Sink, and ACNGS will be the largest BWR in the nation when completed, failure of ECCS function due to Sump Pump water blockage is of particular concern.

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14. Intervenor contends that the design of the Main Steam Line Range Monitor (MSLRM) is not adequate to detect rapid fuel failure in the core of ACNGS. That MSLRM cannot detect rapid fuel failure is evidenced by the failure of this system at <u>Dresden Unit-3</u> in October of 1974 where an estimated 200 fuel rods failed during an operational transient and at <u>Three Mile Island Unit-2</u>, where fuel failure was not detected until 3 hours after the feedwater transient of March 28th. The MSLRM cannot be set low enough to detect rapid fuel failure with sufficient sensitivity due to the presence of N^{16} which gives false positive alarms unless the monitor is set too high to detect rapid fuel failure. NUREG-0401 (March 1978) concludes that a BWR flow blockage accident resulting in fuel failure cannot be detected with certainty by the MSLRM. This intervenor argues:
   (a) Since the MSLRM does function without N^{16} interference in a Pressurized Water Reactor(PWR) that nuclear steam supply system should be constructed at ACNGS, or:
   (b) Applicant should consider the use of a Gamma Dose Rate Alarm, employing an Ionization Chamber (See: <u>Health Physics</u>, 36, Feb., 1979, 195-99).

The higher power core density and high total output thermal power of ACNGS, make rapid fuel failure a higher probability than any other functioning BWR, and thus the contention is suitable for the construction license hearing.

15. Intervenor contends the Lattice Physics Model (See: part 4.3.3 of Sup#2 of SER) is conservative to an "industry standard" WIGLE code that is <u>not</u> conservative in describing the neutron effects of the nuclear design. WIGLE failed in the unsafe direction indicating it underestimated the energy of a power excursion accident (PEA) by 50% (See: <u>IN-1370</u>, p. 87). This Intervenor contends his health interest is endangered by a PEA if Applicant is permitted to install fuel and reactivity control systems based on the Lattice Physics Model due to these findings and also beause the smaller fuel rod size and greater power core density increase reactivity effects such as fissioning and heating. Further, Intervenor contends Applicant should

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be required to install fuel and reactivity control systems that are in keeping with the above findings by having smaller output per fuel rod and fewer rods per fuel assembly.

16. Intervenor contends during Anticipated Transient without SCRAM (ATWS), Power Coolant Mismatch Accident (PCMA), delayed SCRAM or other transient causing events, the fuel rods will cause steam blanketing of the Emergency Core Cooling System (ECCS) coolant, preventing them from being adequately cooled, leading to fuel melt with consequent explosion danger from a molten metal-water reaction or inability of the reactor to cool a pile of fuel which has melted and fallen to the bottom of the reactor vessel. Further, a primary research objective of the Loss of Fluids Test (LOFT) facility appears (NUREG-0006, March, 1979) to be to understand heat transfer such as this during a cooling emergency, even though the major tests on adequacy of ECCS used depressurized fuel rods. Intervenor concludes his health and safety interest may be impaired by steam blanketing of the fuel rods and that this reactor with its extremely high power core density and high thermal output makes the problem a particularly great hazard.

17. Intervenor contends pressure from blowdown following a power excursion accident (PEA), Loss of Coolant Accident (LOCA) or Power Coolant Mismatch Accident (PCMA) combined with a single or several stuck relief valves may hit the suppression pool with sufficient force to permit escape of radio-active gases by causing cracks in the containment building wall and endanger Intervenor's health and genetic safety interests. Stuck relief valves have been a source of "Reportable Occurences", most recently at <u>Three Mile Island, Unit-2</u>, where closure failure led to a severe accident. Moreover, in <u>Nuclear Safety</u> 19(1) p. 82,,20% of "Reportable Occurences or 246 for BWR listed equiptment, involved valves for 1976 and 20% of these or 241 were reported in 1977 (<u>Nuclear Safety</u>, 20(1), p. [illegible]4). These two volumes listed 51 pressure relief system reports for 1976 and 74 for 1977. NUREG-0462 (July 1978) listed 97 relief valve failures in 10 years with BWRs, including 27 failures to open during operation and 17 during testing. This indicates considerable unreliability in pressure relief systems in BWRs, and possible

deterioration with use or age in reliability. This Intervenor contends Applicant must provide relief valves made by several manufacturers and installed alternatively to avoid a common mode failure of manufacture which would result in uneven distribution of blowdown pressure to the suppression pool.

18. In the event the main steam line valve trips, or the turbine trips and the SCRAM fails, the proposed plant whould have a power excursion accident (PEA) if the recirculation pumps could not be tripped rapidly. In-core pressure sensors which are relied upon to trip the recirculation pumps are not in a position where they can be tested for their response to dangerous high pressure, except during refueling and these sensors have high rates of "Reportable Occurence". Summarizing, 1976 for 22 BWRs, <u>Nuclear Safety</u>, 19(1) P. 82, shows 128 occurences, and summarizing 1977, the same journal (20(1), p.84) shows 123 occurence reports for 23 BWRs. This journal shows 246 reportable occurences for valves in 1976 and 241 for 1977. The high level of reportable occurences indicates this system is not reliable in its sensing and its operation aspects. In view of the high power core density proposed for the ACNGS BWR/6 reactor the unreliability of the recirculation pump trip system is a hazard to petitioners health and safety interest. Applicant should be required to show it has researched the previous experience of other licensees to determine which manufacturer's products are the most reliable, and to present possible redundant systems to alleviate these deficiencies.

19. Intervenor contends Applicant's control rod drive system (CRDS) is susceptible to cracking collet retainer tubes which may prevent operation of two or more control rod drives during an accident sequence, making safe shutdown impossible thus endangering Intervenor's health and safety. Twenty-four (24) of sixty-five (65) inspected CRDs were found cracked in 1975 at the <u>Monticello</u> nuclear plant. NUREG-0479 reports that 10% of 779 collet retainer in 11 BWRs had such cracks. While it is expected Applicant will comply with several changes recommended by its vendor due to the end of cycle reactivity effect, such changes will not alter the

contended situation. This Intervenor contends Applicant should be required to present evidence the system proposed is superior to the one currently in use with BWRs in respect to collet retainer tube cracking, taking into account the high power core density, large number of control rods and other design parameters of ACNGS.

20. Fuel performance calculations using GEGAP-III plus fission gas correction factor for burnups greater than 20,000 megawatt-days per ton of uranium, have only been determined by Applicant's vendor to bound the BWR/5 system with regard to the only effected safety system, the Emergency Core Cooling System (ECCS). With a BWR/6, Intervenor contends significant alteration of the ECCS or use of a BWR/5 may be required because the maximum temperature, 1,821°F., reached in the calculation with a BWR/5, is but 379°F. short of the LOCA limit. The BWR/6 design is untested. Intervenor contends the BWR/6 is unready for use because Applicant's vendor has not determined if these calculations clearly apply to a BWR/6, and that these should be accomplished before the operating licese hearing at a special hearing or at the construction license hearing. Further, should the results indicate large measures need to be taken, then a BWR/5 should replace the BWR/6 design. This is particularly important because of ACNG's large thermal output power. Intervenor contends this change from usual procedure and change in applicant's BWR design is required by the health and economic risk of inadequate ECCS.

21. Intervenor contends changes in the technical specifications of ACNGS because of the resolution of the amount of reactivity inserted during collapse of voids due to an overpressure transient should not be placed in the operating license stage. This is because 10 CFR 50.27 permits the "substantial completion" of the plant before the license is issued which risks Applicant's facility being too far along for modification or for modification without expensive changes. In the former, only derating would be possible if the reactivity inserted due to the collapse of voids were found larger than expected. In the latter there would be the aforementioned expense. In either case Intervenor's

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economic interest would be effected. Although General Electric has issued its topical report, the Staff's consultant questions the results, so Intervenor's contention has a basis.

22. Intervenor contends the control rods may develope cracking in the blades in-core which hold the neutron absorbing boron carbide, B_4C. Since the centermost rods are subject to the greatest neutron absorbing requirement, these may crack severally, resulting in an amount of B_4C greater than the highest rod worth being dispersed in the coolant and resulting in inability to shutdown. Intervenor allges an economic interest because prolonged inadequate shutdown will cause loss of use of the plant, and likely strain Applicant's system which is not interconnected with any utility sending power out of the state. Further, the use of the 8 x 8 (minus one) fuel assembly makes the issue relevant to ACNGS because the control rods each have to absorb more neutrons than with a 7 x 7 fuel assembly.

23. The design based loss of coolant accident (LOCA) assumes a large pipe break due to a fault in the pipe while the reactor is operating at full power. However, the accident could also occur by being initiated by the pressure surge due to an overheating transient such as a power excursion or coolant flow blockage. Intervenor contends his health and safety interests are not protected by the design based LOCA, because of the accident possibility cited and that the Emergency Core Cooling System (ECCS) should be designed to mitigate such an accident. Applicant's large power core density makes consideration of this problem at the construction license stage appropriate.

24. The design based control rod drop accident mitigation does not protect Intervenor's health, safety and economic interest sufficiently because more radio-activity than calculated will be released and more damage will be done to the reactor by the reactivity initiated by this design based accident. Recently, NEDO- 10,527, "Rod Drop Analysis for BWRs", showed that the 280 calories per gram peak energy yield limit would be exceeded

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if the reactivity worth of the dropped rod was 2.5%. Recalculation in that publication showed 1.4% would yield 280 cal. per gram peak energy yield. However, the control rod worths have not been changed in BWR reactors, such that the 1.4% can be exceeded in the design based accident as is shown in the Montague Nuclear Plant, PSAR. Intervenor contends where ACNGS has a longer active fuel length and a greater power core density, this anomoly is more serious than with the Montague design. Intervenor contends the control rod system must have a greater number of rods with each controlling a smaller number of fuel rods in order to bring the reactivity worth of each rod to no more than 1.4% and the peak energy yield to less than 280 cal. per gram of fuel.

25. The design based accident for a flow blockage incident is inadequate because it assumes blockage of but one fuel assembly. The only known accident of this type (a very serious one) involved blockage of two fuel assemblies (Fermi, Unit-1, 1966). In addition, NUREG-0401, "Fuel Failure Detection in Operating Reactors," March 1978, Page 23, concludes that, ". . . a possible exception (to adequacy of fuel failure detection) is associated with a postulated BWR flow blockage accident which might proceed undetected in its early stages." In-core and in-reactor matierials and parts may work loose due to corrosion or mechanical failure including internal parts of piping, and fuel assembly support parts such as intermediary spacers, and channel box portions. Steam dryer section could do this also. Where ACNGS is planned to produce more thermal energy than any BWR to date in a configuration of reactor internals largely untried at full scale, the danger of this accident is greater than the smaller 7 x 7 fuel design plants. This Intervenor contends there is DANGER in such an accident of severe economic loss and possible danger to health and safety if fuel melts as in the Fermi Unit-1 accident. Applicant should be required to develope data on an accident involving additional flow blockage than the

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design based accident and incorporate any new developments which improve flow blockage detection reported to be in "experimental stages" in NUREG-0401.

26. The reactor "stud bolts" may give way leading to a severe accident. Applicant has stated in the PSAR that since the moderating water is unboarated, the bolts do not need inspection or other safety procedures. Intervenor contends bolts should be visually inspected at refueling time because the consequences of containment vessel head partial lift-off or other failure presents a high risk to his safety because the reactor vessel fission products would be released into the containment building where they would be difficult to cool. If the head blew off, and pierced containment building roof, then obviously this would be worse. Where the proposed ACNGS will have such a high thermal power output, the accident liklihood is greater than other BWRs.

27. Intervenor contends that the pedestal concrete of ACNGS may be weakened by the heat from a power excursion accident (PEA) or loss of coolant accident (LOCA) such that restart and operation of the reactor would endanger Intervenor's health and safety through subsequent reactor movement due to the original thermal damage to the pedestal. Subsequent damage would include but not be limited to pipe breaks and other accidents costing much to repair and releasing radio-activity in excess of 10 CFR 50, Appendix I. Damage occured to the Dresden Units II and III, in 1971 during drywell over-pressurization to which three General Electric Engineers testified at Senate Hearings in 1976, and a small reactor tore loose from its pedestal in Idaho in 1961 (SL-1) following a power excursion, (See: Thompson, T. J.. and Beckerly, J. G., Technology of Nuclear Reactor Safety, V. 1, (1964), p. 676) and injuring the concrete. Further, the exposure of pedestal concrete to thermally extreme conditions occured at Three Mile Island, Unit-2, in March 1979, and must be expected as a possibility for ACNGS. This Inter-

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venor contends the proposed ACNGS should be constructed to withstand long term thermal shock to its reactor pedestal in view of the probable impossibility to evaluate Three Mile Island, Unit-2 for a considerable time, and Applicant should file such plans for Staff evaluation before the construction license hearings, to protect public safety and avoid expensive retrofit. Since ACNGS is far more thermally powerful than the reactors that have been subject to pedestal damage the appropriate forum is these hearings.

28. Applicant's PSAR does not consider the possibility of a control rod ejection accident adequately. This endangers Intervenor's interests with a reactivity insertion accident. Applicant states (P. 15.1-75 of the PSAR) that, "In all cases the subsequent withdrawal speeds (that is speeds due to an 'unplanned withdrawal' which are variable) are less than a rod drop accidant", infering the consequences must be less. But, in a control rod ejection, the rod would be forced out by the containment pressure and possibly the pressure in the SCRAM discharge volume tank (SDVT) would be additional, creating a more rapid rod ejection as opposed to rod drop. That such SDVT pressures have occured is set forth in TexPIRG's contention #32 of May 16, 1979. Rapid pulling of a rod, led to a fatal power excursion with the Stationary Low Power Plant (S-L-1) reactor in January, 1961. Applicant should be required to show its control rod system is safe for a control rod ejection accident against transients, calculate the effects of a control rod ejection on the public safety and show why a control rod ejection is impossible.

29. Applicant's heat sink has unexplainably been reduced to insufficient size, and that, "there is insufficient assurance that postulated failures would not lead to unacceptable blockage of the submerged intake canal," (Sup #2 SER). Enlargement is needed to protect the reactor from thermal destruction during and following a Loss of Coolant Accident (LOCA) or other accident where a large quantity of cool water is needed for a long

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period of time. These insufficiencies present a risk of meltdown and release of radio-activity harmful to Intervenor's health and safety and risk of reactor core destruction which would result in economic loss to this Intervenor as well. The basis of this contention is the Staff's position. This Intervenor contends:

(a) the ultimate heat sink must be of greater volume and constructed to prevent blockage of the intake canal; and:

(b) that weekly inspections are inadequate monitoring of passive systems as was revealed by the failure of inspection of liquid waste storage tanks at Hanford, Washington, in 1973, (See: <u>Science</u>: 24 August, 1973, <u>181</u>, No. 4101);and:

(c) that research conclusions in NUREG/CR-0003, (Feb. 1978 p. 4),indicate, "...more data and analysis are required for useful understanding of cooling pond thermal performance".

30. Intervenor contends his health, safety and economic interests are imperiled because Applicant refuses to interconnect with any utility which interconnects with an out of state utility. This refusal makes ACNGS safety systems more vulnerable to lack of power in the event there is loss of off-site power during severe climactic conditions or other disturbances, and the proposed unit must turn on on-site diesel generators which are not highly reliable, and unpreferred to the use of off-site power. Further, in July, 1974, the <u>Vermont Yankee</u> BWR experienced a turbine trip due to lightning during severe climactic conditions and in September, 1977, the <u>Donald C. Cook - Unit 1</u>, had the same event. Applicant's grid may become too highly centered around ACNGS because it will produce a high proportion of the power and will be a base load operating plant particularly during non-peak hours and non-peak seasons. Applicant should be required to show that interconnection is not necessary to provide grid stability and adequate power to the ACNGS safety systems without use of the unpreferred power source, with the condition that if

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this cannot be shown, Applicant will be required to interconnect with out-of-state interconnected utilities before a construction license is granted. Or, alternatively Applicant should be required to either:

(a) shut down ACNGS when there are severe climactic conditions; or:

(b) provide a third generator and generator start-up when severe climactic conditions occur.

31. Intervenor contends coolant flow induced vibration of the fuel assemblies will lead to degradation of the Local Power Range Monitors (LPRMs) signal due to wear or other damage, to the extent reactivity monitoring and control in several significant fuel rods will become unreliable, exceeding the ± 5.4% error in Radiation Monitoring Systems and leading to administrative derating of the reactor. Intervenor contends Applicant should provide additional LPRMs to give additional information on the BWR core's power characteristics sufficient to prevent either administrative derating or accident hazards such as power excursions. Current plans for 3 LPRMs are not sufficient.

32. Intervenor contends **the vaporization** rate of the Emergency Core Cooling System (ECCS) water during a design based loss of coolant accident (LOCA) is more rapid than the General Electric (GE) model predicted for an 8 x 8 fuel assembly core such as ACNGS. This was reported by the Advisory Committee on Reactor Safeguards from its March 8-10, 1979 evaluation meeting on the <u>William Zimmer Nuclear Power Station, Unit 1</u> from data collected by GE using the two loop test apparatus. Further, Intervenor contends:

(a) any revision of the ECCS model should be used in determining the capacity requirements for the ECCS of ACNGS, and:

(b) a calculation should be made to include the fact the ACNGS channel boxes are of different thickness than the enclosures of the <u>Zimmer</u> plant, and:

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(c) the resulus of the revision and calculation should be applied in the construction of the ECCS of ACNGS.

33. Applicant's fuel design and reactivity control system relies too much on reactivity and pwer excursion control by the Doppler effect. Applicant's reactor manufacturer, General Electric(GE) relies on experiments that used metalic fuel in the test cores instead of uranium oxide fuel. Metalic uranium fuel gives up its fission heat much more rapidly than oxide fuel such that the coolant heating effect (Doppler effect) which would reduce the moderating ability of the coolant and decrease fissioning is less in the ACNGS case than the experiments in which the extent of the Doppler effect in mitigation of reactivity initiated accidents or transients was determined (BORAX -1). Oxide fuel tests such as SPERT-1 had waterlogged fuel rods, which diminished their reactivity, and SPERT-3, was never tested for upper limit of Doppler negative reactivity effect, and the current Power Burst Test Facility (PBTF) uses depressurized fuel rods which diminishes reactivity when inserted. Intervenor contends there is no experimental data to show the Doppler effect will function to protect Intervenor's health interests against a severe explosion as much as GE and Applicant believe. Further, ACNGS is the largest BWR proposed and hence the effect of overestimate of Doppler reactivity feedback will be more severe and more likely than with any previous nuclear power facility in NRC jurisdiction. Further, ACNGS has a reactivity control system ("Fast" SCRAM) and an 8 x 8 fuel assembly design. Neither of these is fully understood or as well understood in the effect of Doppler negative reactivity feedback as the earlier reactivity control and fuel assembly systems. Intervenor would have Applicant's design altered by the above considerations by:

(a) Having the reactor system altered to either BWR/1, BWR/2, or BWR/4.

(b) Delaying the construction license until excursion testing can be done on large core reactors with oxide fuel which has been irradiated.

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34. Intervenor's economic interest will be injured if the plant is granted a construction license because the General Electric Corporation (GE) division making nuclear steam supply systems (NSSS) will be going out of business because of business misfortune, and the recent Three Mile Island Unit 2 incident, substantially before (35 years) the licensing period of ACNGS will be complete (the year 2027). Intervenor will be further economically injured if the proposed NSSS is installed because obtaining (1) Parts, (2) information, and (3) qualified repair and advisory personnel from GE will be more difficult as years progress. Such "difficulty" will be experienced as delay, plant outage, NRC ordered deratings, and operating under less safe conditions. As there are few (if any) BWR-6/ Mark-III units in operation or preparation, these three items may also be impossible to obtain without great expense and be of poor quality if GE withdraws. Because of the above, Intervenor contends:

(a) A different NSSS supplier should be obtained by Applicant, or:

(b) Applicant should be required to obtain a contract to supply the above (satisfactory to the Board, and Parties) which contains a clause that in the event GE ceases NSSS production these items will be supplied at costs no higher than that available for such items from NSSS vendors in business at that time.

35. Applicant will be unable to provide safe welding of piping at ACNGS without costly repairs to such welding or danger to petitioners health and economic interests in the event of pipe break as a result of such welding not being rewelded when it should have been. Welding at Commanche Peak Nuclear Steam Station Units 1 & 2 in Somervell County, Texas, has been done frequently by persons being trained to be welders prompting large frequency of rewelding and seven meetings between NRC officials and the utility representatives. This Intervenor says the same situation is likely to occur here due to a shortage of trained employees and less than union wages from Applicant's constructor, Ebasco. Intervenor contends Applicant should be

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required to present a program for training persons before they weld at the ACNGS site, and to require a pay scale for employees of all contractors for welding and welders equal to union wages for welders at similar construction conditions, in order to assure continued employment of such welders.

36. Applicant's failure to include mass transfer effects in the analysis of containment vacuum breaker system malfunction constitutes a health risk to this Intervenor. In the event of inadvertant start of containment spray, pressure outside the drywell would decrease to less than the drywell pressure opening the vacuum breaker valves. Blowdown at this point would permit direct forcing of blowdown steam above the wet well water introducing the possibility of containment building overpressurization, drywell overpressurization and suppression chamber overpressurization. Another source of containment vacuum breaker system malfunction is the valves themselves. Six vacuum breaker valves were found open in 1972 at <u>Quad Cities, Unit-2</u> despite test panel indication they were "closed". Intervenor contends Applicant must commit itself to shutdown and inspection of these valves in event of containment spray operation. In addition, Applicant should analyze all design based accidents taking into account mass transfer effects, and agree to install valves of different manufacture to prevent common mode failure due to manufacture error.

37. Applicant's Emergency Core Cooling System (ECCS) does not take into account the effect of heat in the walls of the reactor in the event of a Loss of Coolant Accident (LOCA) and does not meet General Design Criteria 15 and 36. Research on the "Hot Wall Effect", NUREG/CR-0599 (April, 1979) concludes that experiments at larger scale than those performed to date are needed to confirm the best estimate result of research for dimensionless liquid scaling of conuntercurrent flow, before it is believed all parameters of the ECCS are included. Where ACNGS is the largest BWR system attempted, this lack of research knowledge is likely to be most acutely lacking and result in the largest underesti-

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mate of hot wall effect. Intervenor contends Applicant's ECCS must be of greater capacity than the GESSAR indicates because of the uncertainty of this new finding.

38. Applicant's alternative to a single failure proof Residual Heat Removal System (RHR) is an unnecessary risk to Intervenor's health and economic interest because it involves the use of the critical reactor depressurization system, and an air compressor system or bottled gas pressure system, which is unnecessarily complex. Applicant has not stated why the RHR system cannot be made single failure proof. That valves can be unreliable is shown in NUREG -0462 (July 1978) "Technical Report on Operating Experiences with BWR Pressure Relief Systems." There have been 97 pressure relief valve failures in 10 years with BWRS, including 27 failures to open during operation, 17 "potential failures" to open, and 53 inadvertant openings. For these reasons Intervenor contends Applicant should be required to design a single failure proof RHR system.

SERVICE OF PROCESS

Copies of "John F. Doherty's Additional Contentions after ALAB-535" were served via U. S. Postal Service, this 25 th day of May, from Houston, Texas, to the parties below. Those marked with an asterisk were hand delivered. Twenty (20) conformed copies were sent to Mr. Chase Stephens, Docketing and Service, NRC.

Sheldon J. Wolfe, Esq. (NRC)
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Respectfully Submitted,

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