OCT 21 1960

Dear Dr. Silverman:

Under date of September 21, 1960 the General Manager's office transmitted to the Advisory Committee on Reactor Safeguards for comment a draft of criteria for the evaluation of sites for power and testing reactors which was developed by the Division of Licensing and Regulation.

In your response of September 26 you state that the Committee believes that that draft could be developed into a useful technical contribution to reactor safety studies, but that there are a number of reasons why the Committee cannot recommend that it be given the status of a Commission regulation. You further state that in the near future you will send us a memorandum on site criteria which sets forth the Committee's views on the matter.

The staff has been instructed to bring a draft of site criteria to the Commission for its consideration before the end of November. It would be extremely helpful if the Commission could have the benefit of the Committee's further views before the staff completes work on the draft criteria. Accordingly, we would appreciate your sending us your memorandum as soon as you can conveniently do so.

This is a difficult and complex problem and the Commission appreciates the time and effort which the Committee has been devoting to it.

Sincerely yours,

c.c.: Chairman (2)
GM (1)
AGMR&S (1)
Yellow record - Mr. Price, DL&R

(Signed) John A. McCone

Chairman

Dr. Leslie Silverman
Chairman, Advisory Committee on Reactor Safeguards to the U. S. Atomic Energy Commission
Washington 25, D. C.

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Appendix B

Conservatisms in the Assumptions and Factors Used in Calculating the Consequences of the Maximum Credible Accident

1. The probability and consequences of catastrophic reactor accidents have been the subject of widespread interest and study since the earliest days of reactor development. To date, however, the technology has not progressed to the point where it is possible to assign quantitative numbers to all the significant factors relative to safety or to predict with surety the probabilities of malfunctioning of engineering features of plant design under all operating conditions that might exist. There is rather general agreement, however, as expressed in the Brookhaven Report (AEC Report WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants), that the probability of a major accident in reactor plants as we know them today is exceedingly small. The following is quoted from the report:

> "As to the probabilities of major reactor accidents, some experts believe that numerical estimates of a quantity so vague and uncertain as the likelihood of occurrence of major reactor accidents has no meaning. They decline to express their feeling about this probabillity in numbers. Others, though admitting similar uncertainty, nevertheless ventured to express their opinions in numerical terms .... However, whether numerically expressed or not, there was no disagreement in the opinion that the probability of major reactor accidents is exceedingly low."

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use 4 periods instead of dashes,

2. This low probability of occurrence is due to both the inherently safe features of reactors and the safeguards that have been engineered into the plants as a part of deliberate and planned effort to insure safety.

3. The conservatism reflected in the reactor plants is revealed through the analytical technique of postulating a severe accident condition and then evaluating the ability of the plant to remain under control and, through the safeguards provided, including location, prevent or minimize the effects of release of hazardous radioactive effluents. Whereas the exact probability of a major release cannot be predicted, it is possible to arrive at a judgment on site suitability through analysis of the conservatism reflected both in design and the assumptions made in calculating the consequences of a major accident. This in brief is the general approach that has been used by the AEC and the ACRS to arrive at their judgments on applications for construction permits.

4. The "maximum credible accident" is defined as that accident, usually an imaginatively postulated one, which would result in the most hazardous release of fission products; the potential hazard from this accident would not be exceeded by that of any other accident whose occurrence during the lifetime of the facility would appear to be credible.

5. For pressurized and boiling water reactors, for example, the maximum credible accident has been postulated as the complete loss of coolant upon complete rupture of a major pipe, with consequent expansion of the coolant as flashing steam, meltdown of the fuel and partial release of the fission product inventory to the atmosphere of the reactor building.

6. Power and testing reactors presently being operated or under construction near inhabited areas, pursuant to licenses issued by the Commission, are enclosed within external containment vessels. This outer barrier to fission product release to the atmosphere has within its enclosure all or a substantial part of the primary plant coolant piping systems representing

an inner barrier. Cladding on the fuel provides an additional barrier that acts as a retaining "can" for the fissionable material and the fission products formed. Thus, gross release of fission products to the atmosphere woul only occur after the breeching of two inner barriers: the fuel/cladding and the primary system, and then the external "barrier of last resort," the containment building.

7. The manner by which this might be initiated must follow one of two processes. First, through uncontrolled energy release to the confined coolant to produce pressure enough to rupture the coolant piping; or through mechanical failure of the piping or pressure retaining barrier. In either case loss of the coolant would set the stage for possible fuel meltdown from the decay nuclear heat.

8. The rupture of the coolant system from high internal pressures due to uncontrolled internal heat generation requires that:

    (1) Reactivity control mechanisms fail to function and,

    (2) High-pressure relief systems fail to perform,

    (3) Pressures exceed rupture limits of the piping material.

These prior failures need not occur for the case of a <u>spontaneous</u> pipe rupture. However, for such a case, the assumption of a complete shear of a pipe represents an extremely unlikely event. Nevertheless, assuming that such a break should occur and coolant is lost, fuel melting requires that:

a/ Decay heat is sufficient to increase fuel temperature to the melting point;

f. Safeguard systems provided to flood or spray the core with water are either inoperative or insufficient to keep fuel temperatures from rising.

9. Despite such safeguards as those described above, if a major release of fission products to the environment should occur, estimations of the exposure doses which might result to persons offsite are extremely difficult to make because of the complex and interwoven technical effects involved. Although the amount of each kind of radioactive material present in a reactor system can be estimated fairly closely, as a function of the power level history, how much of this material would be released as a result of an accident is highly unpredictable. Quantities in the order of 10 - 30% of the total inventory have been assumed in the past. Experimental data would indicate these values to be conservative but the exact release can vary so much from reactor system to system and with the detailed nature of an accident that the exact degree of conservatism is not known. Further, there is a multiplicity of possible patterns of atmospheric dispersal whereby these radioactive material can be transported to areas beyond the site boundary and those patterns can vary markedly from one reactor location to another.

10. In accidents of the "maximum credible" type, the radioactive materials, along with erosion and corrosion products, first would be dispersed in the coolant through melting or rupture of fuel elements, then find passage to the outer containment barrier through breeches in the coolant system. On breeching, the further expansion to a larger volume and a lower pressure in the containment vessel results in steam, in addition to the gaseous fission products, and production of aerosols as well as miscellaneous sizes of particulate matters. Some ejected materials may conceivably burn on contact

with air, thus increasing the volatiles and fractions of smaller particles. At the same time, a certain amount of fallout within the reactor building or containment structure might be expected as well as condensation of the steam upon contact with cooler surfaces. The fallout is complicated by conversion of normally gaseous fission products into solids by decay, and condensation of volatiles by cooling. Fallout by diffusion and settling process under gravity is complicated by the agitations of turbulence and convection. Superimposed on these factors is the radioactive decay resulting in reduction of source strength with time by conversion to more stable isotopes. All these factors pose a very difficult problem if one ~~really wishes~~ attempts to determin with any exactness the radioactive content of the air which leaks out of the final barrier (containment vessel).

11. The end objective of estimating this radioactive load within this final barrier is to attain a starting point for calculating the radiation hazard to those in the surrounding environs. For those in close proximity, this container of radioactivity represents a source of direct gamma radiation, attenuated by such factors as the structural shielding, distance, time decay and shielding by the topography. For those at more distant points, the transport by air of the materials leaking from the containment vessel becomes determining. For air transport, factors such as the nature of the material leaking from the containment vessel, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important, and many of these are a function of the area in which the reactor is located.

12. It is from this complexity of interwoven technical parameters that criteria for use in the selection of sites has been formulated.

While these criteria represent a considerable simplification of the many complex phenomena involved, they represent the same very conservative approach to site selection that has characterized such evaluations in the past. The fundamental assumptions upon which the proposed bench mark distances are based with estimates of the degree of conservatism represented in each case are as follows:

a) Experts agree and experience to date, though limited, confirms that there is only an exceedingly small probability of a serious accident in reactors approved or likely to be approved for construction. The probability is still lower for an accident in which significant amounts of fission products are released into the confined primary coolant system; and yet a great deal lower for accidents which would release significant quantities of radioactivity from the primary system into the reactor building.

b) It is assumed that the maximum credible accident ~~shall~~ will release into the reactor building 100% of the noble gases, 50% of the halogens and 1% of the solids in the fission product inventory. This is approximately equal to 15% of the total fission product inventory. (The other 85% remain trapped within the fuel matrix or the plant primary system.)

3) The release of radioactivity from the reactor building to the environment shall be considered to occur at a leak rate of 0.1% per day. It is assumed that the leakage and pressure conditions persist throughout the effective course of the accident, which for practical purposes, is until the iodine activity has decayed away.

The maximum pressure within the reactor building and the leakage would of course decrease with time as the steam condenses from contact with cooling surfaces. By assuming no change in leak rate as a function of pressure drop, a conservative factor of at least 5 - 10 is introduced into final off-site dose calculations.

d 50% settling of particles in the containment vessel is assumed in the bench mark criterion but credit has not been taken for the effects of washdown or filtering from protective safeguards such as cooling sprays and internal air recirculating system.

It is estimated that settling could give an effect of 3 - 10 reduction in the end result. Washdown features and filtering networks could provide additional reduction factors of 10 - 1000.

e Atmospheric dispersion of material from the reactor building is assumed to occur according to a relationship developed by O. G. Sutton involving meteorological factors of wind velocity, atmospheric stability and diffusion parameters. ~~and particle deposition rates.~~

This relationship is representative of the current state-of-the-art for calculating downwind concentrations of dispersed material from a source, though there are other more complex relationships believed to be somewhat more accurate - and less conservative. It has been estimated that the use of the more accurate equations might result in reduction in calculated effects by 3 at distances in the order of 3 miles and a factor greater than 3 at 10 miles.

f) The bench marks assume no shift in wind direction for the duration of the accident.

The effect of assuming wind variability depends upon the pressure reduction rate within the containment vessel. Reductions in the order of 2 - 50 might be realized through wind direction shifts. Wind meandering from any one centerline direction might also result in a reduction factor of approximately 2.

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g) Atmospheric dispersion is assumed to be under inversion type weather conditions. For weather conditions which exist for 75% or so of time at most sites, the atmospheric dispersion conditions would be more favorable, by factors of 5 - 1000.

h) No ground deposition (particulate fallout) is assumed for the evacuation distance.

Deposition during cloud travel could reduce the evacuation distance by factors of 2 - 5.

Thus, there is exceedingly high probability that, even if a maximum credible accident should occur, the resulting exposure doses would be many times lower than those calculated by the proposed bench mark calculations.

13. On the other hand, it must always be remembered that there are potential, conceivable accidents which would involve larger fission product releases than those assumed to be released in the maximum credible accident, and conceivably the consequences could be more hazardous to people. This,

and other potentially more hazardous factors than those represented by the proposed site criteria, include:

a) Total radioactivity releases could theoretically be up to ~~nine~~ six times as large as those assumed.

b) Release of long-lived fission products could theoretically be up to 99 times as large as those assumed. This would have far ranging effects on bone dose exposures and on long term contamination of ground areas.

c) The weather conditions could be worse than those assumed, over a small percentage of the time, increasing exposure doses by a factor of 10 or more.