UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
September 29, 1981

Docket No. 50-244
LS05-81-09-074

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPIC II-1.C, POTENTIAL HAZARDS DUE TO NEARBY TRANSPORTATION, INSTITUTIONAL, INDUSTRIAL AND MILITARY FACILITIES - R. E. GINNA

Enclosed is the staff's final evaluation of SEP Topic II-1.C for the R. E. Ginna Nuclear Power Plant. This evaluation is based on our review of your topic safety assessment report submitted by letter dated April 15, 1981 and supplemented by letter dated August 20, 1981.

This completes our evaluation of Topic II-1.C.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8110020442 810929
PDR ADOCK 05000244
P PDR

Mr. John E. Maier

cc

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Bureau of Nuclear
Operations
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

U. S. Environmental Protection Agency
Region II Office
ATTN: Regional Radiation Representative
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

R. E. GINNA
SYSTEMATIC EVALUATION PROGRAM
II-1.C, POTENTIAL HAZARDS DUE TO
NEARBY TRANSPORTATION INSTITUTIONAL INDUSTRIAL AND
MILITARY FACILITIES

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for hazards originating at nearby facilities.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic III-4.D, "Site Proximity Missiles" reviews the extent to which the facility is protected against missiles originating from offsite facilities.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity."

V. EVALUATION

There is little industrial activity in the vicinity of the Ginna plant. Wayne County, where Ginna is located, is primarily a rural area. Typical industries for Wayne County are shown in Table 2.5-1 of the FSAR, reproduced here as Table 1. The nearest concentration of industrial activity is located in the town of Webster, about 6 miles from the site, and consists primarily of light manufacturing (Xerox copiers). No industrial development is expected to occur in the vicinity of the Ginna site.

The nearest transportation routes to the plant are Lake Road and U. S. Route 104, which pass about 1700 feet and 3 1/2 miles, respectively, from the plant at their closest point of approach.

The guidance of Regulatory Guide 1.91, Revision 1, was utilized to evaluate the consequences of postulated explosions on Lake Road. Regulatory Guide 1.91, Revision 1, has been specifically identified by the NRC's Regulatory Requirements Review Committee as needing consideration for backfit on operating reactors. The highway separation distances at Ginna exceed the minimum distance criteria given in the Regulatory Guide and, therefore, provide reasonable assurance that transportation accidents resulting in explosions of truck-size shipments of hazardous materials will not have an adverse effect on the safe operation of the plant. It is important to note that no hazardous cargo would be expected to be transported along Lake Road. This road is used primarily for local traffic, such as that relating to the apple processing plants. No industry using large quantities of explosives is located along

this route. Any large quantities of hazardous material would be shipped via U.S. Route 104 which, at 3 1/2 miles from the plant site, is sufficiently distant not to be of concern.

Highway accidents on Lake Road involving certain hazardous chemicals could theoretically exceed toxicity limits in the plant control room assuming an optimum set of spill parameters and atmospheric dispersion conditions. However, the highway separation distances and the lack of any indication of frequent shipment of hazardous chemicals past the plant (since shipment would be along U.S. Route 104), provide reasonable assurance that the likelihood of a hazardous chemical spill affecting the operation of the plant is low. This matter is being evaluated separate from SEP under NUREG-0737, Item III.D.3.4, "Control Room Habitability."

The nearest railroad to the plant is the Ontario Midland railroad about 3 1/2 miles to the south. Comparing this distance with the guidance provided in Regulatory Guide 1.91, it is apparent that potential railroad accidents involving hazardous materials are not considered to be a credible risk to the safe operation of the plant.

The nearest large pipelines to the plant are a 12" gas line located about six miles southwest of the plant, and a 16" gas line located about 10 miles south of the plant. These pipelines are far enough removed to assure that pipeline accidents will not affect the safety of the nuclear plant. Figure 1 shows a portion of the residential gas lines serving homes along Lake Road, as well as the house heating boiler at the Ginna plant itself. There are no gas or oil production fields, underground storage facilities, or refineries in the vicinity of the plant.

The potential effect of the gas line service to the Ginna house heating boiler was discussed during the Ginna Fire Protection review. This 4-inch gas line comes into the plant underground until it penetrates the ground surface at the east end of the screenhouse. This routing ensures separation from all other safety-related structures and systems. At this point, a metering station and a gas shutoff valve are located (the gas meter was relocated as a result of the Fire Protection review, item 3.1.13). The gas line is buried underground again after the gas meter regulator station, and enters the building through the basement wall under the house heating boiler area. The gas pipe is of welded steel construction up to the boiler. There is continuous ventilation of the areas that the gas line passes through within the building. The gas line service to the boiler and the boiler controls were reviewed and compared to NFPA-85, as requested in the staff's Fire Protection SER, dated February 14, 1979 (item

3.1.46) and was found acceptable in Supplement No. 2 to the Fire Protection SER, issued on February 6, 1981. Based on the resolution of all gas line items during the Fire Protection review, it can be concluded that no safety hazard results from the existence of the gas line on the plant site.

There are no large commercial harbors along the southern shore of Lake Ontario near the plant. Some freight is shipped through Rochester harbor about 20 miles to the west. Major shipping lanes in the lake are located well off-shore, at least 23 miles or more, from the plant.[7] The possibility of damage to the service water intake structure was also considered. Section III-B.27 of RG&E's "Technical Supplement Accompanying Application for a Full-Term Operating License," August 1972 discusses the design of the intake system. As noted in this report, the intake system is completely submerged below the surface of the lake. A ten-foot reinforced concrete lined tunnel, driven through bedrock, extends 3100 feet northerly from the shoreline. The tunnel rises vertically and connects to a reinforced concrete inlet section. The occurrence of historical low water level will result in a depth of water of 30 feet at the inlet and with 15 feet of cover over the inlet structure. This is sufficient to prevent damage from any boating which might pass in the vicinity of the structure. Further, plugging of inlet water flow by a single large piece of material is prevented by the design of the inlet structure, in that water enters on a full 360° circle.[13] Another design feature

at Ginna to ensure continued availability of essential service water is that service water intake can be directly drawn from the discharge canal, which is located on the plant site, protected from any potential lake boating. Thus, lake navigation is not considered to be a hazard to the plant.

The closest airport to the plant is the Williamson Flying Club Airport, a small privately-owned general aviation facility located approximately ten miles ESE.

The Williamson Flying Club Airport has one paved runway. This runway, designated 10-28 and thus oriented in an almost east-west direction, is 3377 feet long and 40 feet wide. The main runway is equipped with low intensity runway lights. The airport has instrument approach capability to runway 28 from the Rochester VORTAC. Figure 2 shows the instrument flight path. There is no control tower at this airport. The airport is used for general aviation activities such as business and pleasure flying, and for agricultural spraying operations. There are currently about 5,000 operations per year at the facility, and about 30 based aircraft, including part-time based crop dusters. The great majority of the aircraft are single-engine propeller airplanes which typically weigh on the order of 1500 to 3600 pounds.[8]

The small number of operations at this airport is substantially fewer than the criteria given in Section III.3 of SRP 3.5.1.6 and therefore is not considered a potential hazard.

Monroe County Airport, in Rochester, New York, located about 25 miles southwest of the plant, is the nearest airport with scheduled commercial air service. Low altitude federal airways V2 and V2N pass about 10 miles south and 2 1/2 miles southwest of the plant, respectively. The low altitude federal airways, V2 and V2N, serve about 10 flights per day. Almost all flights use V2, with V2N being used only occasionally. At most, 10% of airline traffic would use V2N. The width of these airways are eight miles.[9] We have reviewed the probability for an airline crash from these airways in accordance with the method given in SRP 3.5.1.6 Section III-2. The calculated probabilities are 5.1×10^{-8} for airway V2 and 1.4×10^{-8} for airway V2N. Since both airways probabilities are less than the 1×10^{-7} acceptance criteria, we conclude that the probability of a commercial air traffic crash at Ginna is acceptable.

Air Force Restricted Area R-5203 is located about eight miles north of the plant site. Whenever flight activity is conducted by the Air Force within R-5203, radar surveillance is maintained by the 21st NORAD Region, the 108th Tactical Control Group, or possibly the Cleveland Air Route Traffic Control Center. Pilots rely upon on-board navigational equipment to maintain their presence within the specified limits of the restricted

area. Pilots can also be advised if their aircrafts stray beyond their limits by the radar surveillance unit covering the area at the time. The restricted area is used daily for military flight training which includes high-speed interceptor training maneuvers, operational flight checks, and air-to-air refueling. The current altitude ranges from 2,000 to 50,000 feet above the surface.[5] A portion of the Detroit Sectional Aeronautical Chart, reproduced as Figure 3, shows the airports, air routes, and training space described above. There is also a slow-speed low altitude military training route (SR-826) which passes about 6 miles west of the plant. Acceptance criterion II.2 of SRP 3.5.1.6 states that, for military air space, a minium distance of five miles is adequate for low level training routes, except those associated with unusual activities, such as practice bombing. Air Force Restricted Area R-5203 is about eight miles from at its closest boundary, and no unusual activities such as practice bombing take place. The slow-speed low altitude military training route SR-826 is about 6 miles from the plant. Therefore, this criterion is met.

VI. CONCLUSION

Since current regulatory criteria are met with regard to SEP Topic II-1.C, it can be concluded that this topic is complete for the R. E. Ginna site. No additional review for this topic is required during the SEP integrated assessment.

References

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.5.
2. Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Sections 2.2.1, 2.2.2, 2.2.3, and 3.5.1.6, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Sterling Power Project Nuclear Unit No. 1 Preliminary Safety Analysis Report Addendum, Rochester Gas and Electric, Volume 1, Sections 2.1 and 2.2.
6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, Rev. 1, February 1978.
7. Conversation with Chief, U.S. Coast Guard Station, Rochester, New York, 4/8/81.
8. Conversation with Vern Tyrrell, manager of the Williamson Flying Club Airport, 4/7/81.
9. Conversation with FAA controller, Monroe County Airport, 4/8/81.
10. Fire Protection SER, Dennis L. Ziemann to Leon D. White, Jr., February 14, 1979.
11. Fire Protection SER, Supplement No. 2, Dennis M. Crutchfield to John E. Maier, February 6, 1981.

12. Letter, Dennis M. Crutchfield, NRC, to John E. Maier, RG&E, SEP Topics II-3.A, II-3.B, II-3.B.1, II-3.C, dated April 10, 1981.

13. Rochester Gas and Electric Corporation, "Technical Supplement Accompanying Application for a Full Term Operating License," August 1972.

Table 1

FSAR TABLE 2.5-1

TYPICAL INDUSTRIES IN WAYNE COUNTY

Company and Product	Distance from Site	Direction from Site
National Distillers & Chemical Corp. (Kordite Div.), Macedon Polyethelene Products	14-1/2 mi.	South
Duffy-Mott Co., Inc. Williamson Baby Foods	8-1/2 mi.	Southeast
Garlock, Inc. Palmyra Mechanical Packings	15 mi.	Southeast
Bloomer Bros. Co. Newark, Folding Paper Boxes	19 mi.	Southeast
Jackson Perkins Co. Newark Nurserymen	19 mi.	Southeast
Sarah Coventry, Inc. Newark Direct-mail sales of costume jewelry	19 mi.	Southeast
National Biscuit Co. (Dromedary Co. Div.) Lyons, Cake mixes, dates and peels	19 mi.	Southeast
General Electric Co., Clyde Electronic Equipment	27-1/2 mi.	Southeast
Comstock Foods Inc., Red Creek Canned Foods	31 mi.	East
Kenmore Machine Products, Inc. Lyons Refrigerant Products	22 mi.	Southeast
Olney & Carpenter, Inc. Wolcott Canned Foods	27-1/2 mi.	East
C. W. Stuart & Co. Newark Nurserymen	19 mi.	Southeast
Francis Leggett Co., Sodus Canned Foods	12-1/2 mi.	East
The Waterman Food Products Co. Food Processing	3-4 miles	South
Ontario Kraut Corp. 7 Railroad Ave. Food Processing	3-4 miles	South SW
Victor Preserving Co. Food Processing	3-4 miles	South
Ontario Cold Storage Food Processing	3-4 miles	South SW
Waterman Fruit Products Co. Food Processing	3-4 miles	South SW
Ontario Food Products Food Processing	3-4 miles	South SW
Lyndan Products Co. Food Processing	3-4 miles	South SW

FIGURE 1

Small Gas Lines in the Vicinity of Ginna

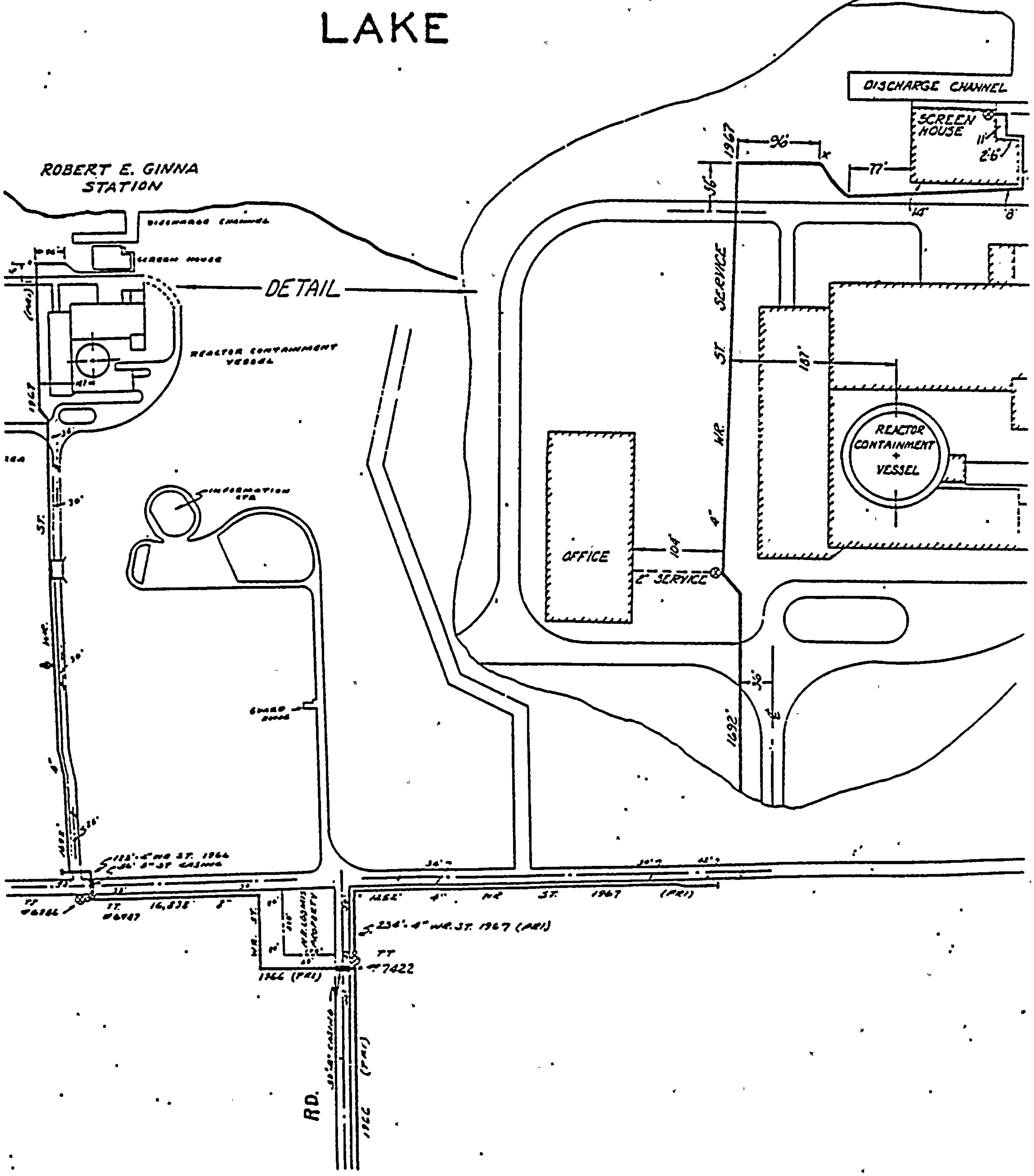
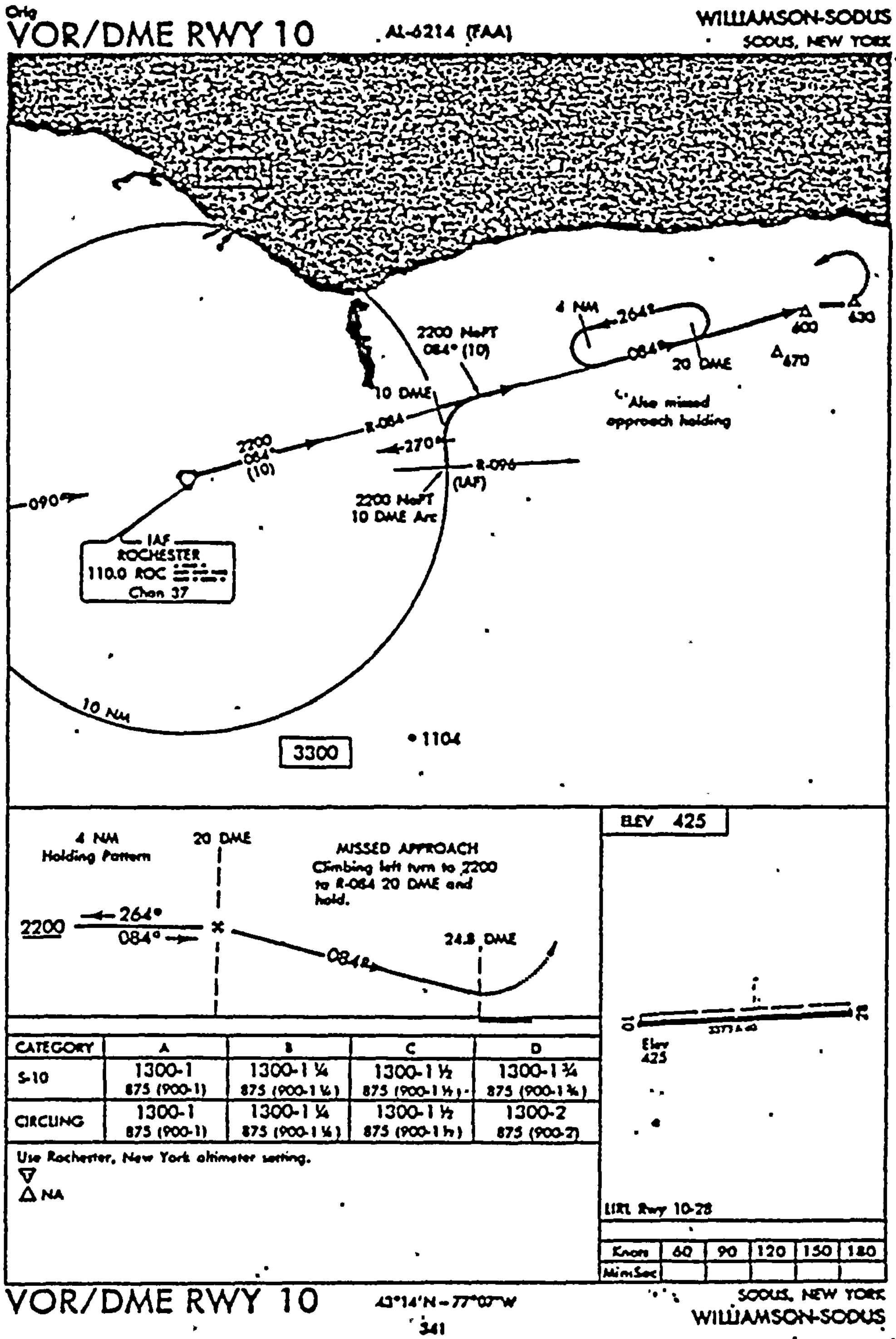
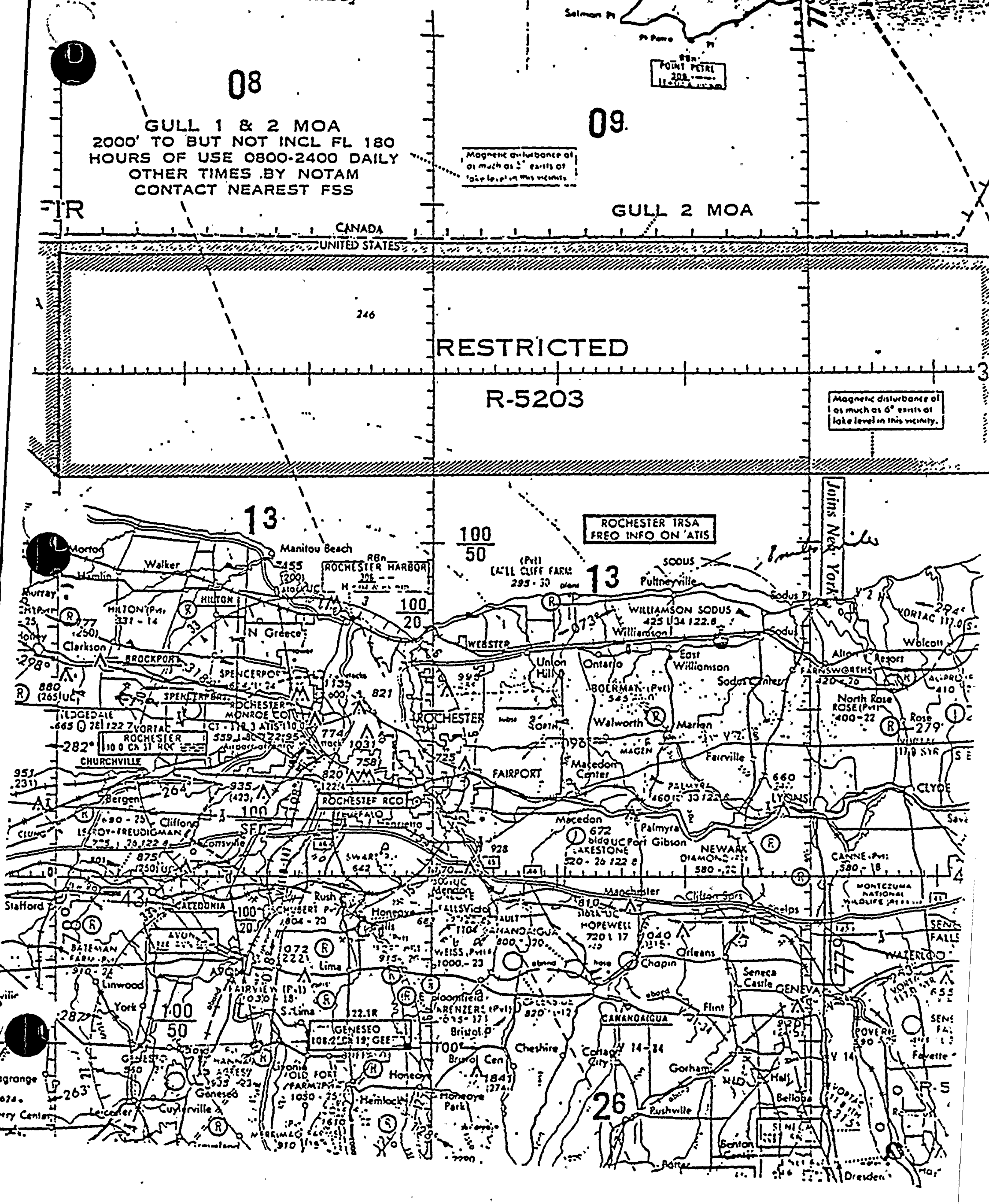


FIGURE 2

Instrument Landing Path to WFC Airport



Detroit Sectional Aeronautical Chart - Rochester Vicinity



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 3, 1981

Docket No. 50-244
LS05-81- 057

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

Subject: SEP TOPIC II-2.A, SEVERE WEATHER PHENOMENA

Enclosed is our final evaluation of SEP Topic II-2.A, "Severe Weather Phenomena" for the R. E. Ginna site. This evaluation (Enclosure 1) incorporates those comments provided in your letter dated January 19, 1981. In accordance with your comments we have provided the appropriate references to our data. In addition, you requested that we provide the references which were not available to you. These are included as Enclosures 2 and 3.

Your letter indicated a concern regarding the snow load provided in our original evaluation. We have reviewed your comments and have revised our estimated snow load as indicated in the evaluation.

Finally, you stated that, on a reasonable design basis, tornado loadings for the Ginna site need not be considered. You further stated that you would evaluate the available data and provide the NRC with a reasonable design basis wind load. Subsequent communications with your staff have indicated that you are not pursuing this matter. Based on your concern, we have reevaluated our design basis tornado and have provided a revised design basis tornado as indicated in our final evaluation.

This evaluation will be a basic input to the integrated safety assessment for your facility. This assessment may be revised in the future if your facility design is changed or if the NRC criteria relating to this subject are modified before the integrated assessment is complete.

Sincerely,

for Thomas V. Wambach

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosures:
As stated

Mr. John E. Maier

cc
Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Bureau of Nuclear
Operations
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ENCLOSURE 1

Systematic Evaluation Program

Meteorology

R. E. Ginna Nuclear Power Plant

Topic II-2.A Severe Weather Phenomena

Extreme meteorological conditions and severe weather phenomena in the Ginna site region were examined to determine if safety-related structures, systems, and components are designed to function under all severe weather conditions. Discussed below are the severe weather phenoma which could adversely affect the Ginna site and which should be examined relative to the current design.

Normal daily temperatures range from a minimum of 18 degrees Fahrenheit in January to a maximum of 82 degrees Fahrenheit in July.[1, 2, 3] Measured extreme temperatures for the site region are 100 degrees Fahrenheit which occurred in June 1953 and -16 degrees Fahrenheit which occurred in February 1961.[4] The extreme maximum and minimum temperatures appropriate at the Ginna site for general plant design (i.e., HVAC systems) are 91 degrees Fahrenheit (equalled or exceeded 1% of the time) and 2 degrees Fahrenheit (equalled or exceeded 99% of the time).[11]

Thunderstorms occur an average of 29 days per year in the site region. Based on the annual number of thunderstorm days, the calculated annual flash density of ground lightning strikes is four flashes per square kilometer.[10] A structure with the approximate dimensions of the Ginna reactor building can be expected to be subjected, on the average, to one strike every 10 years.

The design wind speed (defined as the "fastest-mile" wind speed at a height of 30 feet above ground level with a return period of 100 years) acceptable for the site region is 85 miles per hour.[4,5,6] On the average, hail storms occur about two days annually, and freezing rain occurs approximately 12 days per year.[4] The maximum radial thickness of ice expected in the site region is about 0.75 inch.[12]

Mean annual snowfall in the site region is approximately 86 inches.[19] In the site area, a maximum monthly snowfall occurred in February 1958 and totaled 72.6 inches.[19] The maximum snowfall from a single storm totaled 43.5 inches in March 1900. The maximum measured snow depth on the ground for the site region is 48 inches.[7] Highly localized effects operate to produce snow falls in the Lake Ontario "snow belt" along the south and east shores of the lake. A recent study[21] in the area has shown that snow loads for this section of the lake shore are about 40-50 lbs/ft^2. If we now add the 48 hours probable maximum winter precipitation[7] to this 50 pound value a total of 100 lbs/ft^2 would result.[15] The 100 lb/ft^2 combined snow load is suggested for structural capability assessment at Ginna. The 100 lb/ft^2 should be generally applicable to the site although local drifting on buildings could produce higher loads.

D

Based on actual tornado occurrences in the site region area a "site-specific" design basis tornado (with a probability of occurrence of 10^{-7} per year) can be calculated. For the Ginna site, the characteristics of tornadoes occurring within a 60 mile radius are a maximum windspeed of 250 miles per hour[22], a maximum pressure drop of 1.5 pounds per square inch, and a rate of pressure drop of 0.6 pound per square inch per second.

The tornado wind speed provided is on the order of the upper 95 percentile value. This value is recommended for use in the SEP evaluations since it compensates for the uncertainties inherent in the analysis. These uncertainties are described below.

At the Ginna site, tornadoes/water spouts were not considered even if they had been observed over the water, thereby lowering the number of tornadoes considered and possibly biasing the results. This reduction of conservatism due to counting tornadoes, uncertainties in the use of various factors for the DAPPLE probability analysis, and the fact that the DAPPLE method reflects only the wind speed at which a structure failed which may not be the maximum wind that occurred can lead to underestimates of wind at a given probability level.

Finally, McDonald's (1980) analysis[22] relies only on the detailed study by Abbey and Fujita (1975) of the 1974 "Super Outbreak" of tornadoes during a two-day period. The study has not been expanded to incorporate additional detailed review of subsequent tornadoes in other areas of the country. As a result, the general applicability of this type of analysis is unknown.

References

1. U.S. Department of Commerce, NOAA, "Climates of the States," Vol. 1, 1974.

2. U.S. Department of Commerce, "Climatic Atlas of the United States," June 1968.

3. U.S. Department of Commerce, NOAA, "Climates of the United States," 1973.

4. U.S. Department of Commerce, NOAA, "Local Climatological Data," Rochester, Syracuse, and Buffalo, New York, 1976.

5. H. C. S. Thom, "New Distributions of Extreme Winds in the United States," Journal of the Structural Division, ASCE, Vol. 94, No. ST7, July 1968.

6. "American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," ANSI, A58.1-1972.

7. "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24, and 48 Hours," Hydrometeorological Report No. 33, Washington, D. C., April 1956.

8. James A. Ruffner and Frank E. Baier, "The Weather Almanac," Gale Research Company, 1974.

9. David M. Ludlum, "Weather Record Book," Weatherwise, Inc., 1973.

10. J. L. Marshall, "Lightning Protection," John Wiley and Sons, New York, 1973.

11. "ASHRAE Handbook of Fundamentals," American Society of Heating, Refrigeration and Air Conditioning Engineers, Inc., New York 1976.

12. Paul Tattleman and Irving I. Gringorten, "Estimated Glaze Ice and Wind Loads at the Earth's Surface for the Contiguous United States," Air Force Cambridge Research Laboratories, October 1973.

13. U.S. Housing and Home Finance Agency, "Snow Load Studies," Housing Research Paper No. 19, May 1952.

14. U.S. Naval Weather Service, "World-Wide Airfield Summaries," Vol. VIII, United States of America, Part 4, 1969.

15. Memo from Harold R. Denton (Assistant Director for Site Safety, Division of Technical Review, NRR) to R. R. Maccary (Assistant Director for

Engineering, Division of Technical Review, NRR) dated March 24, 1975, Subject: Site Analysis Branch Position - Winter Precipitation Loads.

16. Memo from Jerry Harbour (Chief, Site Safety Research Branch, Division of Reactor Safety Research, RES) to L. G. Hulman (Chief, Hydrology-Meteorology Branch, Division of Site Safety and Environmental Analysis, NRR) dated August 14, 1978, Subject: Tornado Frequency Data for SEP Review.

17. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C.

18. WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," U.S. Atomic Energy Commission, May 1974.

19. Sterling Power Project - Nuclear Unit 1 (SNUPPS), Preliminary Safety Analysis Report, Docket No. 50-485.

20. H. C. S. Thom, "Tornado Probabilities," Monthly Weather Review, October-December 1963, pp. 730-736.

21. Steyaert, L.T. el al, "Estimating Water Equivalent Snow Depth from Related Meteorological Variables" National Oceanic and Atmospheric Administration, NUREG/CR-1389, U. S. Nuclear Regulatory Commission, Washington D.C. May 1980.

22. McDonald, J.E., "Tornado and Straight Wind Hazard Probability at the Ginna Nuclear Power Reactor Site New York," Institute for Disaster Research, Texas Tech. University, Lubbock, TX, May 1980.

Reference 15

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 2 4 1975

R. R. Maccary, Assistant Director for Engineering, TR

SITE ANALYSIS BRANCH POSITION - WINTER PRECIPITATION LOADS

To resolve the inconsistencies among applications in the selection of meteorological conditions and recurrence intervals acceptable as bases for normal and extreme winter precipitation loads, we are establishing the following interim position on winter precipitation loads to be included in the load combinations specified in Section 3.8 of the Standard Review Plan. This interim position will be replaced by a Regulatory Guide on extreme meteorological conditions.

Winter precipitation loads to be included in the combination of normal live loads will be based on the weight of the 100-year snowpack or snowfall, whichever is greater, recorded at ground level.

Winter precipitation loads to be included in the combination of extreme live loads will be based on the addition of the weight of the 100-year snowpack at ground level plus the weight of the 48-hour Probable Maximum Winter Precipitation (PMWP) at ground level for the month corresponding to the selected snowpack. Modifications to this procedure may be necessary for certain areas where it can be satisfactorily demonstrated that the PMWP could neither fall nor remain entirely on top of the antecedent snowpack and/or roofs. These modifications will be reviewed on a case-by-case basis.

Snowpack and snowfall should be adjusted for density differences, and all ground-level values should be adjusted to represent appropriate weights on roofs of safety class structures.

A currently acceptable procedure for establishing base 100-year snowpack and snowfall would be to use Figure 4 in Section 7 of ANSI A58.1 (1972) with suitable adjustments for local conditions based on examination of representative long-term (e.g. 30 years or more) regional data, and a maximization of water content for snow depth information.

Currently acceptable procedures for converting ground-level snowpacks and snowfalls to represent appropriate roof loads are described in ANSI A58.1, although these procedures are currently under review. The 48-hour PMWP may be determined for most areas from the following Hydometeorological Reports of the U.S. Weather Bureau (now NOAA):

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

No. 33., "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Merdian for areas from 10 to 1000 square miles and duration of 6, 12, 24, and 48 hours" (1956).

No. 36., "Interim Report, Probable Maximum Precipitation in California" (1961), Revised (1969).

No. 43., "Probable Maximum Precipitation, Northwest States" (1966).

Other references are listed in Section 2.4.3 of the Standard Review Plan.

It appears from the SNUPPS application that the extreme live load combination of the weight of the 100-year snowpack plus the weight of the PMWP, without modifications, will be the controlling load for design purposes. However, there may be some areas, such as the northern tier of states where the PMWP is not large, or in the near-south tier of states where the PMWP must be substantially modified and the attendant snowpack may be relatively small, where the normal load with its multiplier of 1.7 would be controlling for design purposes.

Procedures similar to those described here were submitted on the SNUPPS Docket, and found to be acceptable design bases by the applicant and staff.

Harold R. Denton, Assistant Director
for Site Safety
Division of Technical Review
Office of Nuclear Reactor Regulation

cc: S. Hanauer
F. Schroeder
TR AD's
TR BC's
A. Kenneke
Meteorology Section Personnel
Hydrologic Engineering Section Personnel
R. Klecker
D. Eisenhut
S. Varga

Enclosure 3

Reference 16

AUG 14 1978

MEMORANDUM FOR: L. G. Hulman, Chief
Hydrology-Meteorology Branch, DSE

FROM: J. Harbour, Chief
Site Safety Research Branch, RSR

SUBJECT: TORNADO FREQUENCY DATA FOR SEP REVIEW

As requested in the August 1, 1978 memorandum from L. G. Hulman, NRR, to J. Harbour, RES, we are pleased to provide preliminary results of our research contract with the National Severe Storms Forecast Center, NOAA. The objective of this contract is to evaluate past tornado reports in order to assess their intensity and to reevaluate intensities assigned to tornadoes having occurred since 1971.

The data provided include the following:

1. Listing of all reported tornadoes within 125 nautical miles of the selected site from 1950 to 1977 (date, time, latitude, longitude, number of deaths, FPP scale, azimuth and range from selected site on a polar coordinate grid).

2. Plots of initial touchdown points centered at the selected site.

3. Frequency tables: path length vs. path width; path width vs. F-scale; distribution by month and date; distribution by month and hour.

4. Mean path area by month; mean time of occurrence by month.

5. Overall averages of path length, path area, and initial touchdown time.

As you are aware, Dr. Fujita is independently analyzing the tornado base; reconciliation between the Fujita data set and the NSSFC data is currently being performed by Dr. McDonald, Texas Tech University. Preliminary analysis indicates that significant differences may exist in the two data sets, especially at the higher intensity ratings. Final substantial decisions based on these data probably should not be made until the reconciliation between the two sets is completed.

We note with interest that results of our research program may be useful to NRR in general, and to the SEP effort in particular. We would be pleased to provide any further assistance should you so desire.

Original Signed by
Jerry Harbour

Jerry Harbour, Chief
Site Safety Research Branch
Division of Reactor Safety Research

Enclosure: as stated

Contact: R. Abbey
427-4373

cc: L. C. Shao, RES
R. P. Denise, NRR
D. F. Bunch, NRR
L. L. Beratan, SD

DISTRIBUTION:
Subject
Circ
Chron
Harbour, copy
~~Abbey, rf~~

SSRB	SSRB: BC	GRSR:A/D
R.Abbey/ddh	J. Harbour	L. C. Shao
08/14/78	08/14/78	08/ /78

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 0 7 1981

LETTER TO ALL SEP LICENSEES

Gentlemen:

SUBJECT: DELETION OF SYSTEMATIC EVALUATION PROGRAM TOPICS COVERED BY THREE MILE ISLAND NRC ACTION PLAN, UNRESOLVED SAFETY ISSUES, OR OTHER SEP TOPICS

Topics in the Systematic Evaluation Program (SEP), Phase II, that are being implemented as part of the Three Mile Island (TMI) NRC Action Plan, or Unresolved Safety Issues (USIs), or are duplicated in part by another SEP topic, are being deleted from the SEP to minimize duplication. Enclosure (1) is the list of SEP topics being deleted and their related TMI, USI, or other SEP topic reference. Enclosure (2) is the original SEP topic definition and the basis for our determination that the SEP topic review can be deleted from the SEP program. NUREG-0485, the SEP Summary Status Report, will be revised to delete those topics identified in Enclosure (1). Enclosure (3) is the list of SEP topics not included in Enclosure (1) which have previously been identified as generic. The ongoing generic activity related to each of these topics is also identified (i.e., the multi-plant generic activity number and title or the NRR generic activity number and title).

The NRC review of the issues identified in Enclosure (1) will be performed by the staff responsible for the TMI Action Plan item or the Unresolved Safety Issue. The review and implementation of TMI Action Plan items and USIs are being conducted for all operating reactors separate from the SEP program. Since a number of TMI issues, as well as USIs, will be resolved during the same time frame as the completion of our assessments, the staff will consider the status and corrective actions for TMI and USI items and will, to the extent practicable integrate them into our overall assessment. This would assure that corrective actions required as a result of the SEP Integrated Assessment are coordinated to the extent possible with TMI and USI requirements and not unnecessarily impact plants.

The topics identified in Enclosure (3) rely upon the completion of the related generic activity. Many of these related generic reviews are complete and Safety Evaluation Reports have been issued. Each licensee will be informed by separate correspondence of the status of Enclosure (3) topics and what further action, if any, is requested. The results of these generic topic reviews, i.e., Enclosure (3) topics, will be included in the Integrated Assessment.

Sincerely,

Gus C. Lainas

Gus C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

ENCLOSURE 1

SEP TOPICS BEING DELETED

SEP TOPIC NO.	SEP TITLE	TMI, USI, or SEP No.	TITLE
II-2.B	Meteorological Measurements Program	TMI II.F.3 TMI III.A.1	Instruments for Accident Conditions Emergency Preparedness - Short Term
II-2.D	Meteorological Data in Control Room	TMI II.F.3 TMI III.A.1 TMI I.D.1	Instruments for Accident Conditions Emergency Preparedness - Short Term Control Room Design Reviews
III-8.D	Core Supports & Fuel Integrity	USI A-2	Asymmetric Blowdown Loads
III-9	Support Integrity	USI A-12 USI A-7 USI A-24 USI A-46 SEP III-6 SEP V-1	Steam Generator & Reactor Coolant Pump Support Mark I Containment Qual. of Class 1E Equipment Seismic Qualification Seismic Design Considerations Codes and Standards
III-11	Component Integrity	USI A-46 SEP III-6 USI A-2	Seismic Qual. of Equip. in Operating Plants Seismic Design Considerations Asymmetric Blowdown Loads
III-12	Environmental Qual. of Safety Equip.	USI A-24	Qual. of Class 1E Equipment
V-3	Overpressurization Protection	USI A-26	Reactor Vessel Pressure Transient Protection
V-4	Piping & Safe End Integrity	USI A-42	Pipe Cracks in BWRs
V-8	Steam Generator Integrity	USI A-3,4,5	Steam Generator Tube Integrity
V-13	Water Hammer	USI A-1	Water Hammer

SEP TOPIC NO.	SEP TITLE	TMI, USI, or SEP No.	TITLE
VI-2.A	Pressure-Suppression Type BWR Containments	USI A-7	Mark I Containment
VI-2.B	Subcompartment Analysis	USI A-2	Asymmetric Blowdown Loads
VI-5	Combustible Gas Control	TMI II.B.7 USI A-48	Analysis of Hydrogen Control Hydrogen Management
VI-7.E	ECCS Sump Design	USI A-43	Containment Emergency Sump Performance
VI-8	Control Room Habitability	TMI III.D.3.4	Control Room Habitability
VII-4	Effects of Failure in Non-Safety Related Systems on ESF	USI A-47 USI A-17	Safety Implications of Control Systems Systems Interaction
VII-5	Instruments for Radiation and Process Variables During Accidents	TMI II.F.1 TMI II.F.2 TMI II.F.3	Additional Accident Instrumentation Inadequate Core Cooling Instruments for Accident Conditions
IX-2	Overhead Handling Systems (cranes)	USI A-36	Control of Heavy Loads Near Spent Fuel Pool
X Asi	Auxiliary Feedwater System	TMI II.E.1.1	Auxiliary Feedwater System Evaluation
XIII-1	Conduct of Operations	TMI I.C.6 TMI III.A.1 TMI III.A.2	Correct Performance of Operating Activit Emergency Preparedness - Short Term Emergency Preparedness - Long Term

Enclosure 1

SEP TOPIC No.	SEP TITLE	TMI, USI, or SEP No.	TITLE
XV-21	Spent Fuel Drop Accidents	USI A-36	Control of Heavy Loads Near Spent Fuel Pool
XV-22	Anticipated Transients Without Scram	USI A-9	Anticipated Transients Without Scram
XV-23	Tube Failures in Steam Generators	USI A-3,4,5 USI A-9	Steam Generator Tube Integrity Anticipated Transients Without Scram
24) XV-24	Loss of all AC Power	USI A-44	Station Blackout

Enclosure 1

DEFINITION

TOPIC: II-2.B Onsite Meteorological Measurements Program

1. Definition:

   To review the onsite meteorological measurements program to determine the extent that the licensee complies with 10 CFR Part 50, Appendix E, and Appendix I.

2. Safety Objective:

   To assure that adequate meteorological instrumentation to quantify the off-site exposures from routine releases is available and maintained.

3. Status:

   Onsite meteorological measurements programs are being reviewed as a part of the Appendix I evaluations.

4. References:

   1. 10 CFR 50, Appendix E, Appendix I
   2. R. G. 1.97, Rev. 1
   3. R. G. 1.23
   4. SRP Section 2.3.3

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

   a. TMI Task Action Plan - NUREG 0660 - Task II.F.3, Instrumentation for Monitoring Accident Conditions

   Task II.F.3 requires that appropriate instrumentation be provided for accident monitoring with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Condition During and Following an Accident," issued December 1980, contains the required meteorological instrumentation to quantify the off-site exposure.

   b. TMI Task Action Plan - NUREG 0660 - Task III.A.1, Improve Licensee Emergency Preparedness - Short Term

   Task III.A.1 requires the evaluation of 10 CFR Part 50, Appendix E backfit requirements in accordance with NUREG 0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plan and Preparedness in Support of Nuclear Power Plants." Backfit requirements include review of the Onsite Meteorological Measurement Program.

   The evaluation required by Task II.F.3 and III.A.1 are identical to SEP Topic II-2.B; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: II.2.D Availability of Meteorological Data in the Control Room

1. Definition:

Data from the onsite meteorological program should be available in the control room.

2. Safety Objective:

To assure that the licensee has appropriate meteorological logical data displayed in the control room to assess conditions during and following an accident to allow for: (1) early indication of the need to initiate action necessary to protect portions of the off-site public; and (2) an estimate of the magnitude of the hazard from potential or actual accidental releases.

3. Status:

No work currently being done on this subject for operating plants.

4. References:

1. 10 CFR 50, Appendix E, Appendix I
2. R. G. 1.97, Rev. 1
3. R. G. 1.23
4. SRP Section 2.3.3

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. TMI Task Action Plan - NUREG-0660 - Task II.F.3

Task II.F.3, "Instrumentation for Monitoring Accident Conditions" requires that appropriate instrumentation be provided for accident monitoring with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required. Regulatory Guide 1.97, Revision 2 "Instrumentation for Light-Water Coolant Nuclear Power Plants to Assess Plant and Environs Conditions during and Following an Accident," issued December 1980, contains the required meteorological instrumentation to quantify the off-site exposure.

b. TMI Task Action Plan - NUREG-0660 - Task III.A.1

Task III.A.1, "Improve Licensee Emergency Preparedness—Short Term" requires the evaluation of 10 CFR Part 50, Appendix E backfit requirements in accordance with NUREG 0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plan and Preparendess in Support of Nuclear

Power Plants". Backfit requirements include review of the Onsite Meteorological Measurement Program.

c. TMI Task Action Plan - NUREG-0660 - Task I.D.1

Task I.D.1, "Control Room Design Reviews" requires that operating reactor licensees and applicants for operating licenses perform a detailed control room design review to identify and correct design deficiencies. This review will include an assessment of control room layout, the adequacy of the information provided, the arrangement and identification of important controls and instrumentation displays, the usefulness of the audio and visual alarm systems, the information recording and recall capability, lighting, and other considerations of human factors that have an impact on operator effectiveness.

The evaluations required by Tasks II.F.3, III.A.1 and I.D.1 are identical to SEP Topic II-2.D; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: III-8.D Core Supports and Fuel Integrity

1. Definition:

Abnormal loading conditions on the core supports andd fuel assemblies due to seismic events or LOCAs could cause fuel damage due to impact between fuel assemblies and upper and lower grid plates or lateral impact between fuel assemblies and the core baffle wall. The resulting damage could result in loss of coolable heat transfer geometry, make it impossible to insert control rods, or cause releases of radioactive materials due to fuel pin failure.

2. Safety Objective:

To assure that all credible loading conditions on core supports and fuel assemblies will not result in unacceptable fuel damage or distortion.

3. Status:

DOR is currently reviewing the dynamic loads imposed on the fuel assemblies during a LOCA. Independent analyses are being conducted by staff consultants.

4. References:

1. ASME Section III

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-2, Asymmetric Blowdown Loads on Reactor Primary Coolant System, NUREG-0649

USI A-2 requires that an analysis be performed by licensees to assess the design adequacy of the reactor vessel supports and other structures to withstand the loads when asymmetric LOCA forces are taken into account. The staff has completed its investigation and concluded that an acceptable basis has been provided in NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981, for performing and reviewing plant analyses for asymmetric LOCA loads. The structural acceptance criteria specified in NUREG-0609 are as follow:

The structural integrity of the primary system including the reactor pressure vessel, RPV internals, primary coolant loop, and components must be evaluated against appropriate acceptance criteria to determine if acceptable margins of safety exist. Allowable limits and appropriate loading combinations are set forth in standard review plans (SRPs), which are listed in the table that follows. The staff recognizes that in some specific cases, where "as built" designs are being reevaluated for asymmetric LOCA loads, these design limits may be exceeded. Acceptance of alternative allowable limits will be based on a case-by-case evaluation of the safety margins.

Load combination criteria in general were not addressed as part of this study. Currently the staff requires that seismic (SSE) and LOCA response be combined, along with responses due to other loading as specified by the SRP. An acceptable method for combining elastically generated seismic and LOCA responses is provided in NUREG-0484. Acceptable methods for combining response generated by an inelastic LOCA analysis and elastic seismic analyses will be evaluated on a case-by-case basis.

Item	SRP
Reactor pressure vessel	3.9.3
Reactor internals	3.9.5, 3.9.1
Primary coolant loop piping	3.9.3
ECCS piping	3.9.3
RPV, SG, pump supports	3.8.3
Biological shield wall	3.8.3
Steam-generator compartment wall	3.8.3
Neutron-shield tank	3.8.3

Since USI A-2 also requires the investigation of seismic and LOCA response be combined, the evaluation required by USI A-2 is identical to SEP Topic III-8.D; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: III-9 Support Integrity

1. Definition:

Review the design, design loads, and materials integrity including corrosion and fracture toughness and the inservice inspection programs of supports and restraints including bolting for the reactor vessel, steam generator, reactor coolant pump, torus and other class 1, 2 and 3 safety related components and piping systems.

2. Safety Objective:

To assure adequate support and/or restraint of safety related systems and components under normal and accident loads so that they will not be prevented from performing their intended functions because of support failures.

3. Status:

DOR has ongoing programs to review component supports. Current emphasis is on primary system supports and on piping system supports and restraints (snubbers).

4. References:

1. ASME Section III
2. Pink Book Generic Topics 3-5 and 3-43

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. USI A-12, Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports, NUREG -0510, 0606

The original scope of USI A-12 was the review of the steam generator and reactor coolant pump supports of pressurized water reactors. However, the staff has expanded the review to include other support structures, such as boiling water reactor (BWR) vessel supports, BWR pump supports, pressurized water reactor (PWR) vessel supports and PWR pressurizer supports (NUREG-0577, Section 1.3). This expanded review will be undertaken in accordance with the guidance of Section 4 of NUREG-0577.

b. USI A-7, MARK I Containment Long-Term Program, NUREG-0649

Support integrity of the Torus is being evaluated under USI A-7. Under this task, a short term program that evaluated Mark I containment has provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA. A longer term program for BWR facilities, not yet licensed, is planned wherein the NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owners group prior to the performance of plant-unique structural evaluations. The Mark I Owners group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

c. USI A-24, Environmental Qualification of Safety Related Equipment, NUREG-0371

Snubber operability and degradation of seals is covered under USI A-24.

d. USI A-46, Seismic Qualification of Equipment in Operating Plants, NUREG-0705

Mechanical snubbers are covered under USI A-46.

e. SEP Topic III-6, Seismic Design Considerations

Snubbers are evaluated for capacity under SEP Topic III-6.

f. SEP Topic V-1, Codes and Standards

Inservice Inspection requirements for supports is covered under SEP Topic V-1, which refers to 10 CFR 50.55a. SEP plants currently have surveillance Technical Specifications on snubbers.

The evaluation required by USI A-12, A-7, A-24, A-46, SEP Topics III-6 and V-1 is identical to the evaluation required by SEP Topic III-9; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: III-11 Component Integrity

1. Definition:

Review licensee's criteria, testing procedures, and dynamic analyses employed to assure the structural integrity and functional operability of safety related mechanical equipment under faulted conditions and accident loads. Included are mechanical equipment such as pumps, valves, fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays.

2. Safety Objective:

To confirm the ability of safety related mechanical equipment having experienced problems to function as needed during and after a faulted or accident condition. The capability of safety related mechanical equipment to perform necessary protective actions is essential for plant safety.

3. Status:

This review is not currently underway in DOR.

4. References:

1. 10 CFR 50.55a
2. 10 CFR 50, Appendix A, GDC 2, 4, 14, 15
3. Standard Review Plan 3.9.2
4. ASME Section III
5. Regulatory Guides 1.20 and 1.68
6. IEEE 344-1975
7. Standard Review Plan 3.9.3

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. USI A-46, "Seismic Qualification of Equipment in Operating Plants" - NUREG-0606, 0705

The component integrity (both structural integrity and functional operability) for safety related mechanical and electrical equipment for all operating plants including SEP plants will be addressed in this new USI (A-46).

b. USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System", NUREG-0649

The assessment of faulted loads for the primary loop are being performed under USI A-2. Further, the assessment of high energy pipe breaks consider the effect of accident loads with regard to jet impingement, pipe whip and other reaction loads.

c. <u>SEP Topic III-6, Seismic Design Considerations</u>

The evaluation of equipment structural integrity under seismic loads will be performed under SEP Topic III-6.

The evaluations required by Tasks USI A-46, A-2, and SEP Topic III-6 are identical to SEP Topic III-11; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: III-12 Environmental Qualification of Safety-Related Equipment

1. Definition:

Safety-related electrical and mechanical equipment that is required to survive and function under environmental conditions calculated to result from a loss-of-coolant accident (LOCA) or a postulated main steam line break (MSLB) accident inside containment must be environmentally qualified. In addition, determine whether environment induced failures of non-safety-related equipment could interfere with the operation of safety equipment. Special attention should be given to the effect of beta radiation on exposed organic surfaces, such as gaskets.

2. Safety Objective:

To assure that the mechanical and Class IE electrical equipment of safety systems have been qualified for the most severe environment (temperature, pressure, humidity, chemistry and radiation) of design basis accidents.

3. Status:

Westinghouse is conducting a verification program which is expected to be completed by the end of 1977 for those plants qualified to IEEE - 323 (1971). The Office of Nuclear Regulatory Research (RES) is sponsoring programs relating to Class IE equipment qualification, the results of which can be utilized to determine the adequacy of the equipment previously qualified.

4. References:

1. NUREG 0153, Item 25, "Qualification of Safety-Related Equipment" December 1976
2. DOR Technical Activities, Category B, Item 34, "Environmental Qualifications of Safety-Related Equipment (Post LOCA)", May 1977
3. DSS Technical Activities, Category A, Item 33, "Qualification of Class IE Safety-Related Equipment", April 1977
4. R. G. 1.89

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

USI A-24, Qualification of Safety-Related Equipment, NUREG-0371, NUREG-0606

The issue identified in reference 1 (NUREG-0153, Item 25) and the review criteria, i.e., R.G. 1.89, are identical to those specified in USI A-24. The Task Action Plan for USI A-24 (NUREG-0371) covers the environmental qualification of both electrical and mechanical safety related equipment.

The evaluation required by USI A-24 is identical to SEP Topic III-12; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: V-3 Overpressurization Protection

1. Definition:

Inadvertent overpressurization of the primary system at temperatures below the nil ductility transition temperature may result in reactor vessel failure during heatup and pressurization. Such overpressure transients are caused by pressure surges when the primary system is water solid. The most severe transients have occurred when a charging pump starts up or inadvertent closing of a letdown valve with a charging pump running. Pressure temperature limits as a function of neutron fluence of the material at the reactor vessel beltline are specified in 10 CFR 50, Appendix G. All PWR licensees have been directed to institute interim administrative procedures to prevent damaging pressure transients and on a longer time scale to provide permanent protection which will probably include hardware changes such as high capacity safety/relief valves.

2. Safety Objective:

To protect the primary system from potentially damaging overpressurization transients during plant pressurization and heatup.

3. Status:

Generic review of all PWR licensee submittals is underway. Criteria for evaluation have been developed and refined by NRR/RES. An effort is being made to complete the review sufficiently early to ensure installation of mitigating systems by the end of 1977.

4. References:

1. NUREG 0138

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-26, Reactor Vessel Pressure Transient Protection (NUREG-0410)

Under USI A-26 licensees were requested to modify their systems and procedures to protect against low temperature overpressurization. All operating PWRs have made these modifications and Safety Evaluation Reports for the SEP plants have been issued.

The evaluation required by USI A-26 is identical to SEP Topic V-3; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: V-4 Piping and Safe End Integrity

1. Definition:

Review the safety aspects that affect BWR and PWR piping and safe end integrity for compliance with 10 CFR Part 50, including fracture toughness, flaw evaluation, stress corrosion cracking in BWR and PWR piping, and control of materials and welding.

2. Safety Objective:

To assure continued piping integrity and compliance with 10 CFR Part 50 and applicable industry codes and standards.

3. Status:

The Engineering Branch, DOR, is conducting an ongoing program that includes the as-needed review of those aspects necessary to ensure the continuing integrity of piping systems important to safety including stress corrosion cracking of BWR colant pressure boundary piping. This program will continue for the life of operating reactors.

4. References:

1. Technical Position, Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping
2. ASME Section XI

5. Basis for Deletion (i.e., related TMI Task USI or other SEP Topic):

a. USI A-42, Pipe Cracks in Boiling Water Reactors, NUREG-0510

The scope of USI A-42 is the study of stress corrosion cracking in BWR piping. NUREG-0313, Rev. 1 "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" is the resolution of USI A-42 and presents staff positions.

b. USI A-10, BWR Feedwater Nozzle Cracking and Control Rod Drive Hydraulics Return Line Nozzle Cracking, NUREG-0649.

c. NRR Generic Activity C-7, PWR System Piping, NUREG 0471

The scope of this activity is the study of stress corrosion cracking in PWR piping. NUREG 0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors" recommends the same corrective actions (pg. 2-12) proposed for BWRs in NUREG 0313, Rev. 1, USI A-42.

The evaluation required by USI A-42 and Task C-7 is identical to the evaluation required by SEP Topic V-4; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: V-8 Steam Generator (SG) Integrity

1. Definition:

Review the safety aspects affecting operation of steam generators including secondary water chemistry, tube plugging criteria, inservice inspection, possibly including a dimensional inspection for proper evaluation of denting, steam generator tube leakage, tube denting, flow induced vibration of steam generator tubes, tube repair, and tube bundle or steam generator replacement.

2. Safety Objective:

To ensure that acceptable levels of integrity of that portion of the reactor coolant pressure boundary made up by the steam generator are maintained in accordance with current codes, standards, and/or regulatory criteria during normal and postulated accident conditions. The integrity of the steam generator is needed to ensure that leakage following a postulated design basis accident will not result in doses to the public in excess of 10 CFR Part 100 guidelines and that the emergency core cooling systems will be able to perform their safety functions.

3. Status:

Review of this topic is being performed by the Division of Operating Reactors. This effort will continue for the life of operating reactors.

4. References:

1. Regulatory Guide 1.83 (Revision 1)
2. Regulatory Guide 1.121
3. 10 CFR 50, Appendix A, GDC 30 and 32
4. Pink Book 3-27

5. Basis for Deletion (i.e., related TMI Task; USI or other SEP Topic):

- USI A-3, A-4, A-5, Westinghouse, Combustion Engineering, and Babcock and Wilcox Steam Generator Tube Integrity, NUREG-0649

The definition of this topic and the references cited are covered by USI A-3, 4 and 5. The evaluation for USI A-3, 4 and 5 is identical to SEP Topic V-8; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: V-13 Water Hammer

1. Definition:

Water hammer events have occured in light water reactor systems. Water hammer events increase the probability of pipe breaks and could increase the consequences of certain events such as the loss of coolant accident. The types of water hammer, the vulnerable systems (for example, containment spray, service water, feedwater and steam) and the safety significance of water hammer have been identified and defined in a staff report of May 1977.

2. Safety Objective:

To reduce the probability of water hammer events that have the potential to lead to pipe ruptures in LWR systems which are needed to mitigate the consequences of accidents or that might increase the consequences of accidents previously analyzed.

3. Status:

Generic review is underway. On March 10, 1977, an interdivisional DOR/DSS technical review group was formed to investigate the water hammer issue and to develop a program for its appropriate consideration in licensing reviews and for operating reactors. Consultant work has been performed by CREARE and Livermore Labs.

4. References:

1. "Water Hammer in Nuclear Power Plants", NRC Staff Report, June 1, 1977
2. "An Evaluation of PWR Steam Generator Water Hammer" by G. B. Wallis, P. H. Rothe, et. al. of CREARE Inc., draft, February 1977.
3. Lawrence Livermore Laboratory "An Investigation of Pressure Transient Propagation in Pressurized Water Reactor Feedwater Lines" (Preliminary) S. B. Sutton, April 15, 1977.
4. NRR Technical Activities, Category A, Item 1, Water Hammer, May 1977.

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic:

- USI A-1, Water Hammer, NUREG-0649

The references cited in this topic were the precursors of USI A-1. The evaluation required for USI A-1 is identical to SEP Topic V-13; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: VI-2.A Pressure-Suppression Type BWR Containments

1. Definition:

BWR pressure-suppression type containments (e.g., Mark I containment) are subjected to hydrodynamic loads during the blowdown phase of a LOCA. Those loads have the potential for damaging the components and structures (wetwell, internal structures, restraints, supports and connected systems) of the containment. During a relief valve blowdown into the suppression pool the wetwell (torus) shell and safety/relief valve restraints may be overstressed. The hydrodynamic loads were not explicitly identified and included in the design of the Mark I pressure-suppression containment.

2. Safety Objective:

To assure that the structural integrity of pressure suppression pool containments is maintained under hydrodynamic loading conditions. It has been determined that the upward forces during the blowdown phase following a LOCA potentially cause the Mark I torus to be lifted, causing failure of connecting systems and supports and leading to loss of the containment integrity. Structural modifications and/or changes in the mode of operation might be necessary to assure adequate safety margins.

3. Status:

Mark I containments are currently evaluated in a two step generic review program: The Short-Term Program (STP), completed May 1977, has focused on the determination of the magnitude and significance of hydrodynamic loads. In the Long-Term Program (LTP), to be completed by late 1978, the design basis loads will be finalized and the capability of the containment to withstand the loads within the original design structural margins will be verified. This verification will be based in part on research results from NRC and industry sponsored programs. As a result of the STP, the staff required that Mark I plants be operated with a drywell to wetwell differential pressure of at least one psi to reduce the vertical loads. In addition some licensees have modified the torus support system for additional safety margin.

4. References:

1. Pink Book - Generic Issues (April 1977)
    a. Mark I Containment - STP Technical Specifications
    b. Mark I Containment Evaluation - STP
    c. Mark I Containment Evaluation - LTP
    d. Mark I Safety/Relief Valve Line Restraints in Torus

2. DOR Technical Activities, Category A, April 1977
   a. Item 2, "Mark I Containment STP"
   b. Item 3, "Mark I Containment LTP"
   c. Item 23, "Mark II Containment"

3. DOR Technical Activities, Category B, May 1977, Item 12, "Assessment of Column Buckling Criteria"

4. DSS Technical Activities, Category A, April 1977, Item 31, "Determination of LOCA and SRV Pool Dynamic Loads for Water Suppression Containments"

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

USI A-7, Mark I Containment Long-Term Program, NUREG-0649

Under this task, a short term program that evaluated Mark I containment has provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA. A longer term program for BWR facilities, not yet licensed, is planned wherein the NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owners group prior to the performance of plant-unique structural evaluations. The Mark I Owners group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

The long term program for USI A-7 will assure that all plants with Mark I containments are able to tolerate, without loss of function, the LOCA induced hydrodynamic loads.

The evaluation required by USI A-7 is identical to SEP Topic VI-2.A; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: VI-2.B Subcompartment Analysis

1. Definition:

The rupture of a high energy line inside a containment subcompartment can cause a pressure differential across the walls of the subcompartment. In the case of a rupture of a PWR main coolant pipe adjacent to the reactor vessel, the subcooled blowdown produces pressure differentials in the annulus between the reactor vessel and the shield wall and also within the reactor vessel across the core barrel. This asymmetric pressure distribution generates loads on the reactor vessel support and on reactor vessel internals on other equipment supports and on subcompartment structures which have not been analyzed previously for most operating reactors.

2. Safety Objective:

To assure that the reactor vessel supports, reactor vessel internals, other equipment supports and subcompartment structures are designed with an adequate margin against failure due to these loads. The failure could result in a loss of ECCS capability.

3. Status:

The staff is reviewing the NSSS vendor and architect engineer design codes used to calculate the loads produced by the asymmetric pressure distribution. Analyses have been completed for a limited number of operating plants. The W TMD code is approved. Bechtel, Gilbert and United Engineering have submitted codes for review.

4. References:

1. Pink Book - Generic Issue, Item 3-5, "Asymmetric LOCA Loads - PWR", April 1977
2. DOR Technical Activities, Category A, Item 32, "Asymmetric LOCA Loads (Reactor Vessel Support Problem)", April 1977
3. DSS Technical Activities, Category A, Item 14, "Asymmetric Blowdown Loads on Reactor Vessel", April 1977
4. DPM Technical Activities, Category A, Item 2, "Reactor Vessel Supports (Asymmetric LOCA Loads from Sudden Subcooled Blowdown), April 1977

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-2, Asymmetric Blowdown Loads on Reactor Primary Coolant System, NUREG-0649

The references cited in this topic were the precursors of USI A-2. The evaluation required for USI A-2 is identical to SEP Topic VI-2.B (see also SEP Topic III-8.D); therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: VI-5 Combustible Gas Control

1. Definition:

Review the combustible gas control system to determine the capability of the system to monitor the combustible gas concentration in the containment; to mix combustible gases within the containment atmosphere; and to maintain combustible gas concentrations below the combustion limits (e.g., by recombination, dilution, or purging). For facilities which share recombiners (portable) between units or sites, determine that the recombiners can be made available within a suitable time. For facilities which utilize purging as a primary means of combustible gas control, determine the radiological consequences of the system operation. Reevaluate hydrogen production and accumulation analysis to consider (1) reduction of Zr/water reaction on the basis of five times the Appendix K calculation amount and (2) potential increases in hydrogen production from corrosion of metals inside containment.

2. Safety Objective:

To prevent the formation of combustible gas explosive concentrations in the containment or in localized regions within containment, following a postulated accident; to assure that the radiological consequences of the system operation are acceptable.

3. Status:

Proposed 10 CFR 50.44 would permit a BWR licensee to propose an alternate combustible gas control system in lieu of inerting. Four such proposals for containment atmosphere dilution (CAD) systems are currently under review, and the COGAP II computer code is being revised to perform the system evaluations.

4. References:

1. Proposed Rule 10 CFR 50.44
2. DOR Technical Activities, Category A, Item 8, "Containment Purge During Normal Operation", April 1977
3. DOR Technical Activities, Category A, Item 14, "Inerting Requirements/CAD", April 1977
4. Branch Technical Position CSB 6-2
5. Standard Review Plan 6.2.5

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. TMI Task Action Plan - NUREG-0660 - Task II.B.7, Analysis of Hydrogen Control

As a result of TMI II.B.7 short and long term rulemaking to amend 10 CFR 50.44 has been initiated. The short term rulemaking (interim rule) requires that all Mark I and Mark II containments be inerted. It also required that the owners of all plants with other containments perform certain analyses of accident scenarios involving hydrogen releases and furnish the staff with a proposed approach for mitigating these hydrogen releases.

The longer term rulemaking will address both degraded core and melted core issues. In the area of hydrogen control it will prescribe requirements that are appropriate for operating plants as well as for plants under construction.

b. USI A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment, NUREG-0705

Under USI A-48 a Task Action Plan has been defined and is being developed that encompasses the concerns in the Definition and the Safety Objective of SEP Topic VI-5.

The evaluation required by TMI II.B.7 and USI A-48 is identical to SEP Topic VI-5; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: VI-7 E ECCS Sump Design and Test for Recirculation Mode Effectiveness

1. Definition:

Following a LOCA in a PWR an emergency core cooling system (ECCS) automatically injects water into the system to maintain core cooling. Initially, water is drawn from a large supply tank. Water discharging from the break and containment spray collects in the containment building sump. When the supply tank has emptied to a predetermined level, the ECCS is switched from the "injection" mode to the "recirculation" mode. Water is then drawn from the containment building sump.

ECC systems are required to operate indefinitely in this mode to provide decay heat removal. Certain flow conditions could occur in the sump, which could cause pump failures. These include entrained air, prerotation or vortexing and losses leading to deficient NPSH.

2. Safety Objective:

To confirm effective operation of ECC systems in the recirculation mode.

3. Status:

Confirmation through pre-operational testing is now required on all CPs. Staff has been accepting scaled tests in lieu of pre-op tests at OL stage. Some plants have required modification to achieve vortex control.

4. References:

1. RFP - Vortex Technology (PWR)
2. Reg. Guide 1.79 para. b(2)

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-43, Containment Emergency Sump Reliability, NUREG-0510, 0660

The definition of this topic and the references cited are covered by USI A-43. The evaluation for USI A-43 is identical to SEP Topic VI-7E; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: VI-8 Control Room Habitability

1. Definition:

   Control rooms in operating plants may not fully comply with General Design Criterion 19. This review should include, but not be limited to, analysis of the control room air infiltration rate, ventilation system isolability and filter efficiency, shielding, emergency breathing apparatus, short distance atmospheric dispersion, operator radiation exposure, and on-site toxic gas storage proximity.

2. Safety Objective:

   To assure that the plant operators can safely remain in the control room to manipulate the plant controls after an accident.

3. Status:

   DOR now reviews control room habitability in operating plants when related licensing actions (e.g., assessment of BWR Containment Air Dilution system post-LOCA radiological impact) require it. DSE has a technical assistance contract with the National Bureau of Standards to measure the control room air infiltration rate at a few operating plants. These measurements will be used to gauge the conservatism of the assumed air infiltration rates currently used by NRC. Some reviews are now in progress for plants we have reason to believe do not meet G. D. Criterion 19 (SONGS-1, Vermont Yankee, St. Lucie).

4. References:

   1. SRP 6.4
   2. 10 CFR 50, Appendix A, GDC 19
   3. "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19", by K. G. Murphy and Dr. K. M. Campe, Proceedings of the Thirteenth AEC Air Cleaning Conference
   4. R. G. 1.78
   5. R. G. 1.95, Rev. 1

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

   - TMI Task Action Plan, NUREG-0737, Task III D.3.4, Control Room Habitability Requirements

   The review criteria required by Task III.D.3.4 (NUREG-0737, pg. 3-197) is identical to the review criteria specified in the Definition and References of SEP Topic VI-8; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: VII-4 Effects of Failure in Non-Safety Related Systems on Selected Engineered Safety Features

1. Definition:

Potential combinations of transients and accidents with failures of non-safety-related control systems were not specifically evaluated in the original safety analysis of currently operating reactor plants. Review the effects of control system malfunctions as initiating events for anticipated transients and also as failures concurrent with or subsequent to anticipated events or postulated accidents initiated by a different malfunction (e.g., the effect of the loss of the plant air system on the plant control and monitoring system). A complete discussion is provided in reference 1.

2. Safety Objective:

To assure that any credible combination of a non-safety-related system failure with a postulated transient or accident will not cause unacceptable consequences.

3. Status:

A technical assistance contract with ORNL for failure mode analyses of control systems was initiated to determine sensitive areas of the plant designs. The results of this program in conjunction with the results of the failure mode and effects analyses for transients and accidents being performed under contract by INEL should provide a basis for any new review and safety requirements.

4. References:

1. NUREG-0153, Item 22, "Systematic Review of Normal Plant Operation and Control System Failures", December 1976
2. Memorandum from V. Stello to R. J. Hart; dated 12/23/76, NRR letter No. 46.
3. DOR Task Force Report on SEP, Appendix B (TFL 118), November 1976
   a. Item 33 "Safety Related Control Power"
   b. Item 34 " Safety Related Instrumentation Power"
   c. Item 56 "Effect of Failure in Non-Safety Related Systems During Design Basis Events"
   d. Item 57 "Loss of Plant Air System (Effect on Plant Control and Monitoring)"
   e. Item 77 "Safety Related Control and Instrument Power"
4. DOT Recommended List of SEP Subjects, Spring 1977 C DOT 102, Item 100z, "Loss of Plant Air System (Effect on Plant Control and Monitoring)

5. Basis for Deletion (i.e., related TMI Task USI or other SEP Topic):

a. USI A-47, Safety Implications of Control System, NUREG-0705, 0606

The issue defined in reference 1 (NUREG-0153, Item 22) is as follows:

In evaluating plant safety, the effects of control system malfunctions should be reviewed as initiating events for anticipated transients and also as failures that could occur concurrently or subsequent to postulated anticipated events (initiated by a different malfunction) or postulated accidents."

The issue defined in USI A-47 is in part as follows:

"This issue concerns the potential for transients or accidents being made more severe as a result of the failure or malfunction of control systems. These failures or malfunctions may occur independently, or as a result of the accident or transient under consideration."

b. USI A-17, System Interaction in Nuclear Power Plants, NUREG-0649, 0606

The purpose of this task is to develop a method for conducting a disciplined and systematic review of nuclear power plant systems, for both process function couplings of systems and space couplings, to identify the potential sources and types of systems interactions that are determined to be potentially adverse.

A report has been developed, "Final Report - Phase 1 System Interaction Methodology Application Program," NUREG/CR-1321, SAND 80-0384 whose objectives are:

1. To develop a methodology for conducting a disciplined and systematic review of nuclear power plant systems which facilitates identification and evaluation of systems interactions that affect the likelihood of core damage.

2. To use the methodology to assess the Standard Review Plan to determine the completeness of the plan in identifying and evaluating a limited range of systems interactions.

The work done under USI A-17 may be useful in the development of USI A-47.

The definition of USI A-47 is identical to that of Topic VII-4; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: VII-5 Instruments for Monitoring Radiation and Process Variables During Accidents

1. Definition:

The adequacy of the instruments for monitoring radiation and process variables during accidents has not been reviewed for conformance with Regulatory Guide 1.97. A generic review is planned to assess the licensee's existing or proposed monitoring instruments during and following accidents to determine the adequacy of their range, response and qualifications, and to determine the sufficiency of the variables to be monitored. Certain instruments to monitor conditions beyond the design basis accidents will also be required in accordance with an RRRC determination (Reference 3).

2. Safety Objective:

To assure that plant operators and emergency response personnel have available sufficient information on plant conditions and radiological releases to determine appropriate in-plant and offsite actions throughout the course of any accident. The instrumentation should also provide recorded transient or trend information necessary for post-accident evaluation of the event. The ability to follow the course of accidents beyond the design basis accidents is also required.

3. Status:

Generic review of instrumentation to follow the course of accidents in operating plants and in all plants now under construction or seeking a construction permit will begin with the issuance of Regulatory Guide 1.97, Revision 1, this year. Submittals describing the facilities' post-accident instrumentation will be obtained from all operating licensees and reviewed by the end of 1978. The implementation of Regulatory Guide 1.97, Revision 1 on operating plants is proceeding independent of the SEP. RRRC has determined that Revision 1 to Regulatory Guide 1.97 should be treated as a Category 2 item (backfit on operating plants on a case by case basis).

4. References:

1. H. G. Mangelsdorf (ACRS) memo of 8/14/73 to L. M. Muntzing (Regulation)
2. L. M. Muntzing (Regulation) memo of 11/1/73 to H. G. Mangelsdorf (ACRS)
3. Proposed Revision 1 to Regulatory Guide 1.97, the Enclosure with the 4/4/77 memo R. B. Minogue (SD) to E. G. Case (NRR)
4. SRP 7.5
5. SRP 7.6
6. SRP 11.5
7. T. A. Ippolito (EICSB) memo of 8/12/74 to Emergency Instrumentation Task Force Members
8. Issue 21, NUREG-0153
9. RRRC Meeting Minutes (January 28, 1977)

5. Basis for Deletion (i.e., related TMI Task USI or other SEP Topic):

TMI Task Action Plan - NUREG-0660, 0737 - Task II.F., Instrumentation and Controls

There are three subtasks under Task II.F. as follows:

a. II.F.1 - Additional Accident Monitoring Instrumentation

b. II.F.2 - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

c. II.F.3 - Instruments for Monitoring Accident Conditions

Specific postions on the required instrumentation for II.F.1 and II.F.2 are in NUREG 0737 and Regulatory Guide (R.G.) 1.97, Revision 2 (December 1980). Instrumentation needed for II.F.3 is also in R.G. 1.97, Revision 2.

The emphasis of TMI Task II.F. is the Monitoring of Radiation and Process Variables; guidance for this relies primarily on R.G. 1.97. This is identical to the review proposed in Topic VII-5; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: IX-2 Overhead Handling Systems - Cranes

1. Definition:

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of PWR and BWR spent fuel storage facilities and inside the reactor building. If a heavy object (e. g., a shielded cask) were to drop on the spent fuel or on the reactor core during refueling, there could be a potential for overexposure of plant personnel and for release of radioactivity to the environment. Review the overhead handling system, including sling and other lifting devices, and the potential for the drop of a heavy object on spent fuel including structural effects.

2. Safety Objective:

To assess the safety margins, and improve margins where necessary, of the overhead handling systems to assure that the potential for dropping a heavy object on spent fuel is within acceptable limits and that the potential radiation dose to an individual does not exceed the guidelines of 10 CFR Part 100.

3. Status:

Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants" was issued for comment in February 1976 and references various industry standards. New applications (CP and OL) are reviewed in accordance with the APCSB Branch Technical Position 9-1 which is identical to Regulatory Guide 1.104.

The review of overhead handling systems of operating reactor facilities is performed on a generic basis and has also been identified as a DOR Technical Activity Category A.

4. References:

1. R. G. 1.104
2. APCSB Branch Technical Position 9-1, "Overhead Handling Systems for Nuclear Power Plants"
3. Pink Book - Generic Issue 3-22, "Fuel Cask Drop Analysis", April, 1977.
4. DOR Technical Activities, Category A Item 50, "Control of Heavy Loads Over Spent Fuel", April 1977

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-36, Control of Heavy Loads Near Spent Fuel, NUREG-0649

The review criteria required by USI A-36 (SRP 9.1.4 and NUREG-0554) are identical to the review criteria specified in the References of SEP Topic IX-2 (BTP 9-1 and RG 1.104); therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: X Auxiliary Feedwater System

1. Definition:

Review the auxiliary feedwater system, associated instrumentation, and connection between redundant systems. The review includes the aspects of pump drive and power supply diversity (e.g., electrical and steam-driven sources), and the water supply sources for the auxiliary feedwater system.

2. Safety Objective:

To assure that the auxiliary feedwater system can provide an adequate supply of cooling water to the steam generators for decay heat removal in the event of a loss of all main feedwater. Older PWR plants may not meet the requirement for pump drive and power supply diversity.

3. Status:

Reviews for new license applications are performed in accordance with the SRP. This topic is not under active review for operating plants.

4. References:

1. SRP, 10.4.9.
2. APCSB BTP 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR Plants"

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- TMI Task Action Plan - NUREG-0660 - Task II.E.1.1, Auxiliary Feedwater System Evaluation

The TMI-2 accident and subsequent investigations and studies highlighted the importance of the auxiliary feedwater (AFW) system in the mitigation of severe transients and accidents. Since then, the AFW systems have come under close scrutiny by the NRC and many improvements have been recommended to enhance the reliability of AFW systems for all plants. The scope of the review outlined in the SEP Topic X definition is identical to the scope of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.E.1.1(2) which requires that each PWR plant licensee:

> "Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance."

The review criteria for the evaluations required by Item II.E.1.1(b) are identical to SEP Topic X; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: XIII-1 Conduct of Operations

1. Definition:

The organization, administrative controls and operating experience will be reviewed. The existing organization and administrative controls will be compared with standard technical specifications and guidance provided in Regulatory Guide 1.8 and 1.33 to determine the adequacy of the staff to protect the plant and to operate safely in routine, emergency, and long-term post-accident circumstances. The plant operating history will be reviewed to assess the combination of staff, operating controls and alarms, and administrative controls, in particular plant procedures, emergency planning and offsite preparedness, to determine whether additional staff, qualifications, or administrative controls will be required for continued safe operation.

2. Safety Objective:

To obtain reasonable assurance that the plant has enough people, with sufficient training and experience, and has administrative controls adequate to specify proper operation in routine, emergency and post-accident conditions.

3. Status:

Most of the older plants have staff members that meet the experience and educational requirements given in ANSI N18.1 - 1971 (endorsed by Regulatory Guide 1.8); however, a comparison against current criteria for the composite staff has not been made. These plants have provided training for subsequent plant staffs and plant experience has in general demonstrated safe design and operation. Operating experience review is ongoing; and has been, in general, favorable. However, an analysis of this experience for trends, common elements, and potential hidden problems has not been systematically performed.

A review of Section VI of operating reactor licensees technical specifications was begun in 1974 using Section VI of STS as a model. As of September 1975 these reviews had been completed and the plants licensed prior to this time had been found to: (1) be acceptable and upgrading was not required, (2) require upgrading of only the reporting requirements, or (3) require improvement to be comparable to the STS model. Plants licensed after September 1975 have been reviewed against the STS model. Further review of Section VI, therefore will not be required.

Emergency plans submitted at the OL stage complied with 10 CFR 50 Appendix E 1970; however, these plans are not consistent with the guidance given in new Regulatory Guide 1.101 Rev. 1 1977.

4. References:

1. R. G. 1.8 and 1.33
2. ANSI N18.1 - 1971
3. ANSI N18.7 - 1972 Revised
4. Standard Technical Specifications, Section VI
5. 10 CFR 50, Appendix E
6. R. G. 1.101 Rev. 1 1977
7. SRP 13.3
8. NUREG 75/111, Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans In Support of Fixed Nuclear Facilities
9. EPA Manual of Protective Action Guides and Protective Action for Nuclear Incidents, September 1975
10. Memorandum of Understanding, NRR and OSP on State and Local Preparedness, March 10, 1977

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. TMI Task Action Plan, NUREG-737, Task I.C.6, Guidance on Procedures for Verifying Correct Performance of Operating Activities

Under TMI Task I.C.6 a review of licensee procedures will be conducted to assure that an effective system of verifying the correct performance of operating activities exist. The purpose of this review is to provide a means of reducing human errors and improving the quality of normal operation. References cited for this review are ANSI Standard N18.7-1972 (ANS 3.2) "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)." These are the same references cited for Topic XIII-1.

b. TMI Task Action Plan, NUREG-0660 and 0737, Task III.A.1, "Improving Licensee Emergency Preparedness - Short Term" and Task III.A.2, "Improving Licensee Emergency Preparedness - Long Term"

Under Task III.A.1 a review of 10 CFR Part 50, Appendix E backfit requirements is being conducted in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The scope of NUREG-0654 covers SRP 13.3 and NUREG 75/111.

R.G. 1.101 has been deleted and has been superceded by an amended Appendix E to 10 CFR Part 50 (45 FR 55410, August 19, 1980). Under Task III.A.2 a review of licensees emergency prepardness plans with respect to amended Appendix E will be conducted in accordance with NUREG-0654.

The evaluations required by TMI Tasks I.C.6, III.A.1 and III.A.2 are identical to SEP Topic XIII-1; therefore, this SEP topic has been deleted.

DEFINITION

TOPIC: XV-21 Spent Fuel Cask Drop Accidents

1. Definition:

Review the potential for spent fuel cask drops, the damage which could result from cask drops, and the radiological consequences of a cask drop from fuel damaged within the cask under conditions exceeding the design basis impact on the cask.

2. Safety Objective:

To assure that the damage to fuel within the casks and radiological consequences resulting from a cask drop are acceptable or that acceptable measures have been taken to preclude cask drops.

3. Status:

Fuel cask drop analysis is a generic item which has been completed on some plants or is presently under review for all other operating reactors.

4. References:

1. SRP Section 15.7.5
2. R. G. 1.25
3. Pink Book

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-36, Control of Heavy Loads Near Spent Fuel, NUREG-0649

The review criteria required by USI A-36 (SRP 15.7.5) are identical to the review criteria specified in the References of SEP Topic IX-2; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: XV-22 Anticipated Transients Without Scram

1. Definition:

Review the postulated sequences of events, analytical models, values of parameters used in the analytical models and the predicted results and consequences of events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram). Analyses of the radiological consequences for these transients will be included. Failure of the reactor to shutdown quickly during anticipated transients can lead to unacceptable reactor coolant system pressures and to fuel damage.

2. Safety Objective:

To assure that the reliability of the reactor shutdown systems is high enough so that ATWS events need not be considered or to assure that the consequences of ATWS events are acceptable, i.e., that the reactor coolant system pressure, fuel pressure, fuel thermal and hydraulic performance, maximum containment pressure and radiological consequences are within acceptable limits.

3. Status:

ATWS is a generic topic currently under review to determine a position for all power reactors. BWR licensees have been requested to install reactor coolant pump trips as a short term program measure. All licensees have submitted descriptions of the applicability of vendor generic ATWS reports for their plants. The schedule for review of class C plants, which includes those plants designated for Phase II of SEP, has not yet been developed.

4. References:

   1. Pink Book
   2. WASH 1270
   3. ACRS
   4. TSAR
   5. SRP Section 15.8 and Appendix

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

-- USI A-9, Anticipated Transients Without Scram, NUREG-0606

The reference cited in this topic, i.e., Pink Book, was the precursor of USI A-9. The evaluation required for USI A-9 is identical to SEP Topic XV-22; therefore, this SEP Topic has been deleted.

DEFINITION

TOPIC: XV-23 Multiple Tube Failures in Steam Generators

1. Definition:

Assess the effects of multiple steam generator tube failures (ranging from leaks to double ended ruptures) as a result of pressure differentials that may occur following a LOCA, steam line break or ATWS events.

2. Safety Objective:

Assure that the reflood of the core following a LOCA is possible and that the radiological consequences following these accidents are within the 10 CFR Part 100 guidelines.

3. Status:

The consequences of multiple tube failures have not been analyzed for any plant at the licensing stage. Work has been done for some operating plants, but ultimate goals have yet to be set.

4. References:

1. Prairie Island Docket
2. Turkey Point Docket
3. Surry #1 and #2 Docket
4. ATWS Report

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

a. USI A-3, 4, 5 "Westinghouse, Combustion Engineering, and Babcock and Wilcox Steam Generator Tube Integrity, NUREG-0649.

Two of the tasks of USI A-3, 4, 5 are as follows:

1. Analyses of LOCA with Concurrent Steam Generator Tube Failures,

2. Analyses of Main Steam Line Break.

The analyses required by these two tasks in USI A-3, 4, 5 covers two of the three events specified in the definition.

b. USI A-9, "Anticipated Transients Without Scram (ATWS)" - NUREG-0606

Pressure differentials resulting from ATWS events have been determined to be no greater than those resulting from main steam line break events (NUREG-0460, Vol. 2, Appendix V). The analysis for ATWS event is, therefore, covered under USI A-3, 4, 5.

DEFINITION

TOPIC: XV-24 Loss of All AC Power

1. Definition:

Review plant systems to determine that following loss of all AC power (on and offsite) the reactor is shutdown and core cooling can be initiated. Loss of all AC power causes loss of most emergency equipment and instrumentation.

2. Safety Objective:

To assure that with only DC power, i.e., equipment design, diversity, and operator action are sufficient to initiate core cooling within a short time period (typically 20 minutes).

3. Status:

Not an explicit SRP topic. Availability of some AC power is assumed in all accident/transient analyses. Topic may be considered as an auxiliary fuel pump or RCIC pump diversity spinoff.

4. References:

5. Basis for Deletion (i.e., related TMI Task, USI or other SEP Topic):

- USI A-44, Station Blackout, NUREG-0606

The problem description of USI A-44 is identical to the definition of SEP Topic XV-24, and the review of USI A-44 would be the same as Topic XV-24; therefore, this SEP topic has been deleted.

ENCLOSURE 3

GENERIC SEP TOPICS

SEP TOPIC NO.	SEP TITLE	NRR GENERIC NO.	GENERIC TASK TITLE
~~III-7.B~~ /III-7.A (Note 1)	Inservice Inspection - Containments	B-49 B-38 ✓	Inservice Inspection Criteria for Containments Tendon Surveillance - Bechtel Containments
III-8.A	Loose Parts & Core Barrel Vibration Monitoring	B-60 B-73 C-12	Loose Parts Monitoring System Monitoring for Excessive Vibration Inside Reactor Vessel Primary System Vibration Assessment
V-1	Compliance with Codes & Standards	A-01 A-14	10 CFR 50.55a(g) - ISI 10 CFR 50.55a(g) - Inservice Testing
V-7 (Note 2)	Reactor Coolant Pump Overspeed	B-68	Pump Overspeed during a LOCA
VI-6	Containment Leak Testing	A-04 A-23	Appendix J - Containment Leak Testing Containment Leak Testing
VI-7.D	Long Term Cooling Passive Failures	B-11	Flood of Equipment Important to Safety
VII-1.B	Trip Uncertainty & Setpoint Analysis Review	NUREG-0138 Issue 13	Instrumentation Setpoint Drift
VIII-1.A	Degraded Grid Voltage	A-35 B-23	Adequacy of Offsite Power Systems Degraded Grid Voltage
IX-6	Fire Protection	B-02	Fire Protection
XI-1	Appendix I	A-02	Appendix I - ALARA
XI-2	Radiological Monitoring Systems	A-02 B-67	Appendix I - ALARA Effluent & Process Monitoring Instrumentation

SEP TOPIC NO.		SEP TITLE	NRR GENERIC NO.	GENERIC TASK TITLE
XIII-2		Safeguards/Industrial	A-03	Security Reviews
XVII		Operational QA Program		Annual IE Inspections
III-4.B	(Note 3)	Turbine Missiles	B-46	Analysis of Turbine Disc Cracks
IV-3	(Note 4)	BWR Jet Pumps Operating Indications	B-28	BWR Jet Pump Flow Indication Elimination
VI-7.A.2	(Note 5)	Upper Plenum Injection	D-05	Plant UPI Model Problem
VI-7.A.4	(Note 6)	Core Spray Nozzle Effectiveness	D-12	Non-Jet Pump BWR Core Spray Perform
VII-6		Frequency Decay	A-35	Adequacy of Offsite Power Systems

NOTES:

1. Applies to Palisades and Ginna only, which have prestressed concrete containments. Topic is deleted for all other SEP plants.

2. Applies to Ginna, Haddam Neck, Palisades, San Onofre, and Yankee Rowe, which are PWRs. Topic is deleted for all BWR SEP plants.

3. Applies to Palisades, Ginna, San Onofre, Yankee Rowe, and Haddam Neck. Topic is deleted for all other SEP plants.

4. Applies to Millstone 1 and Dresden 2. Topic is deleted for all other SEP plants.

5. Applies to Ginna only. Topic is deleted for all other SEP plants.

6. Applies to Big Rock Point, Dresden 1, Dresden 2, Millstone 1 and Oyster Creek. Topic is deleted for all other SEP plants.

Enclosure 3

RECEIVED OCT 5 1981

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
September 29, 1981

Distr. to Ginna
NSARB
Response instructions:
R.C. merely to respond
if necessary
Maier
10-5-81

Docket No. 50-244
LS05-81-09-074

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPIC II-1.C, POTENTIAL HAZARDS DUE TO NEARBY TRANSPORTATION, INSTITUTIONAL, INDUSTRIAL AND MILITARY FACILITIES - R. E. GINNA

Enclosed is the staff's final evaluation of SEP Topic II-1.C for the R. E. Ginna Nuclear Power Plant. This evaluation is based on our review of your topic safety assessment report submitted by letter dated April 15, 1981 and supplemented by letter dated August 20, 1981.

This completes our evaluation of Topic II-1.C.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8110020442 13PP.

cc

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Stanley B. Klimberg, Esquire
General Counsel
New York State Energy Office.
Agency Building 2
Empire State Plaza
Albany, New York 12223

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Thomas E. Murley,
Regional Administrator
Nuclear Regulatory Commission, Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

U. S. Environmental Protection Agency
Region II Office
ATTN: Regional Radiation Representative
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Jay Dunkleberger
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

R. E. GINNA
SYSTEMATIC EVALUATION PROGRAM
II-1.C, POTENTIAL HAZARDS DUE TO NEARBY TRANSPORTATION INSTITUTIONAL INDUSTRIAL AND MILITARY FACILITIES

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for hazards originating at nearby facilities.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic III-4.D, "Site Proximity Missiles" reviews the extent to which the facility is protected against missiles originating from offsite facilities.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity."

V. EVALUATION

There is little industrial activity in the vicinity of the Ginna plant. Wayne County, where Ginna is located, is primarily a rural area. Typical industries for Wayne County are shown in Table 2.5-1 of the FSAR, reproduced here as Table 1. The nearest concentration of industrial activity is located in the town of Webster, about 6 miles from the site, and consists primarily of light manufacturing (Xerox copiers). No industrial development is expected to occur in the vicinity of the Ginna site.

The nearest transportation routes to the plant are Lake Road and U. S. Route 104, which pass about 1700 feet and 3 1/2 miles, respectively, from the plant at their closest point of approach.

The guidance of Regulatory Guide 1.91, Revision 1, was utilized to evaluate the consequences of postulated explosions on Lake Road. Regulatory Guide 1.91, Revision 1, has been specifically identified by the NRC's Regulatory Requirements Review Committee as needing consideration for backfit on operating reactors. The highway separation distances at Ginna exceed the minimum distance criteria given in the Regulatory Guide and, therefore, provide reasonable assurance that transportation accidents resulting in explosions of truck-size shipments of hazardous materials will not have an adverse effect on the safe operation of the plant. It is important to note that no hazardous cargo would be expected to be transported along Lake Road. This road is used primarily for local traffic, such as that relating to the apple processing plants. No industry using large quantities of explosives is located along

this route. Any large quantities of hazardous material would be shipped via U.S. Route 104 which, at 3 1/2 miles from the plant site, is sufficiently distant not to be of concern.

Highway accidents on Lake Road involving certain hazardous chemicals could theoretically exceed toxicity limits in the plant control room assuming an optimum set of spill parameters and atmospheric dispersion conditions. However, the highway separation distances and the lack of any indication of frequent shipment of hazardous chemicals past the plant (since shipment would be along U.S. Route 104), provide reasonable assurance that the likelihood of a hazardous chemical spill affecting the operation of the plant is low. This matter is being evaluated separate from SEP under NUREG-0737, Item III.D.3.4, "Control Room Habitability."

The nearest railroad to the plant is the Ontario Midland railroad about 3 1/2 miles to the south. Comparing this distance with the guidance provided in Regulatory Guide 1.91, it is apparent that potential railroad accidents involving hazardous materials are not considered to be a credible risk to the safe operation of the plant.

The nearest large pipelines to the plant are a 12" gas line located about six miles southwest of the plant, and a 16" gas line located about 10 miles south of the plant. These pipelines are far enough removed to assure that pipeline accidents will not affect the safety of the nuclear plant. Figure 1 shows a portion of the residential gas lines serving homes along Lake Road, as well as the house heating boiler at the Ginna plant itself. There are no gas or oil production fields, underground storage facilities, or refineries in the vicinity of the plant.

The potential effect of the gas line service to the Ginna house heating boiler was discussed during the Ginna Fire Protection review. This 4-inch gas line comes into the plant underground until it penetrates the ground surface at the east end of the screenhouse. This routing ensures separation from all other safety-related structures and systems. At this point, a metering station and a gas shutoff valve are located (the gas meter was relocated as a result of the Fire Protection review, item 3.1.13). The gas line is buried underground again after the gas meter regulator station, and enters the building through the basement wall under the house heating boiler area. The gas pipe is of welded steel construction up to the boiler. There is continuous ventilation of the areas that the gas line passes through within the building. The gas line service to the boiler and the boiler controls were reviewed and compared to NFPA-85, as requested in the staff's Fire Protection SER, dated February 14, 1979 (item

3.1.46) and was found acceptable in Supplement No. 2 to the Fire Protection SER, issued on February 6, 1981. Based on the resolution of all gas line items during the Fire Protection review, it can be concluded that no safety hazard results from the existence of the gas line on the plant site.

There are no large commercial harbors along the southern shore of Lake Ontario near the plant. Some freight is shipped through Rochester harbor about 20 miles to the west. Major shipping lanes in the lake are located well off-shore, at least 23 miles or more, from the plant.[7] The possibility of damage to the service water intake structure was also considered. Section III-B.27 of RG&E's "Technical Supplement Accompanying Application for a Full-Term Operating License," August 1972 discusses the design of the intake system. As noted in this report, the intake system is completely submerged below the surface of the lake. A ten-foot reinforced concrete lined tunnel, driven through bedrock, extends 3100 feet northerly from the shoreline. The tunnel rises vertically and connects to a reinforced concrete inlet section. The occurrence of historical low water level will result in a depth of water of 30 feet at the inlet and with 15 feet of cover over the inlet structure. This is sufficient to prevent damage from any boating which might pass in the vicinity of the structure. Further, plugging of inlet water flow by a single large piece of material is prevented by the design of the inlet structure, in that water enters on a full 360° circle.[13] Another design feature

at Ginna to ensure continued availability of essential service water is that service water intake can be directly drawn from the discharge canal, which is located on the plant site, protected from any potential lake boating. Thus, lake navigation is not considered to be a hazard to the plant.

The closest airport to the plant is the Williamson Flying Club Airport, a small privately-owned general aviation facility located approximately ten miles ESE.

The Williamson Flying Club Airport has one paved runway. This runway, designated 10-28 and thus oriented in an almost east-west direction, is 3377 feet long and 40 feet wide. The main runway is equipped with low intensity runway lights. The airport has instrument approach capability to runway 28 from the Rochester VORTAC. Figure 2 shows the instrument flight path. There is no control tower at this airport. The airport is used for general aviation activities such as business and pleasure flying, and for agricultural spraying operations. There are currently about 5,000 operations per year at the facility, and about 30 based aircraft, including part-time based crop dusters. The great majority of the aircraft are single-engine propeller airplanes which typically weigh on the order of 1500 to 3600 pounds.[8]

The small number of operations at this airport is substantially fewer than the criteria given in Section III.3 of SRP 3.5.1.6 and therefore is not considered a potential hazard.

Monroe County Airport, in Rochester, New York, located about 25 miles southwest of the plant, is the nearest airport with scheduled commercial air service. Low altitude federal airways V2 and V2N pass about 10 miles south and 2 1/2 miles southwest of the plant, respectively. The low altitude federal airways, V2 and V2N, serve about 10 flights per day. Almost all flights use V2, with V2N being used only occasionally. At most, 10% of airline traffic would use V2N. The width of these airways are eight miles.[9] We have reviewed the probability for an airline crash from these airways in accordance with the method given in SRP 3.5.1.6 Section III-2. The calculated probabilities are 5.1×10^{-8} for airway V2 and 1.4×10^{-8} for airway V2N. Since both airways probabilities are less than the 1×10^{-7} acceptance criteria, we conclude that the probability of a commercial air traffic crash at Ginna is acceptable.

Air Force Restricted Area R-5203 is located about eight miles north of the plant site. Whenever flight activity is conducted by the Air Force within R-5203, radar surveillance is maintained by the 21st NORAD Region, the 108th Tactical Control Group, or possibly the Cleveland Air Route Traffic Control Center. Pilots rely upon on-board navigational equipment to maintain their presence within the specified limits of the restricted

area. Pilots can also be advised if their aircrafts stray beyond their limits by the radar surveillance unit covering the area at the time. The restricted area is used daily for military flight training which includes high-speed interceptor training maneuvers, operational flight checks, and air-to-air refueling. The current altitude ranges from 2,000 to 50,000 feet above the surface.[5] A portion of the Detroit Sectional Aeronautical Chart, reproduced as Figure 3, shows the airports, air routes, and training space described above. There is also a slow-speed low altitude military training route (SR-826) which passes about 6 miles west of the plant. Acceptance criterion II.2 of SRP 3.5.1.6 states that, for military air space, a minium distance of five miles is adequate for low level training routes, except those associated with unusual activities, such as practice bombing. Air Force Restricted Area R-5203 is about eight miles from at its closest boundary, and no unusual activities such as practice bombing take place. The slow-speed low altitude military training route SR-826 is about 6 miles from the plant. Therefore, this criterion is met.

VI. CONCLUSION

Since current regulatory criteria are met with regard to SEP Topic II-1.C, it can be concluded that this topic is complete for the R. E. Ginna site. No additional review for this topic is required during the SEP integrated assessment.

References

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.5.
2. Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Sections 2.2.1, 2.2.2, 2.2.3, and 3.5.1.6, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Sterling Power Project Nuclear Unit No. 1 Preliminary Safety Analysis Report Addendum, Rochester Gas and Electric, Volume 1, Sections 2.1 and 2.2.
6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, Rev. 1, February 1978.
7. Conversation with Chief, U.S. Coast Guard Station, Rochester, New York, 4/8/81.
8. Conversation with Vern Tyrrell, manager of the Williamson Flying Club Airport, 4/7/81.
9. Conversation with FAA controller, Monroe County Airport, 4/8/81.
10. Fire Protection SER, Dennis L. Ziemann to Leon D. White, Jr., February 14, 1979.
11. Fire Protection SER, Supplement No. 2, Dennis M. Crutchfield to John E. Maier, February 6, 1981.

12. Letter, Dennis M. Crutchfield, NRC, to John E. Maier, RG&E, SEP Topics II-3.A, II-3.B, II-3.B.1, II-3.C, dated April 10, 1981.

13. Rochester Gas and Electric Corporation, "Technical Supplement Accompanying Application for a Full Term Operating License," August 1972.

Table 1

FSAR TABLE 2.5-1

TYPICAL INDUSTRIES IN WAYNE COUNTY

Company and Product	Distance from Site	Direction from Site
National Distillers & Chemical Corp. (Kordite Div.), Macedon Polyethelene Products	14-1/2 mi.	South
Duffy-Mott Co., Inc. Williamson Baby Foods	8-1/2 mi.	Southeast
Garlock, Inc. Palmyra Mechanical Packings	15 mi.	Southeast
Bloomer Bros. Co. Newark, Folding Paper Boxes	19 mi.	Southeast
Jackson Perkins Co. Newark Nurserymen	19 mi.	Southeast
Sarah Coventry, Inc. Newark Direct-mail sales of costume jewelry	19 mi.	Southeast
National Biscuit Co. (Dromedary Co. Div.) Lyons, Cake mixes, dates and peels	19 mi.	Southeast
General Electric Co., Clyde Electronic Equipment	27-1/2 mi.	Southeast
Comstock Foods Inc., Red Creek Canned Foods	31 mi.	East
Kenmore Machine Products, Inc. Lyons Refrigerant Products	22 mi.	Southeast
Olney & Carpenter, Inc. Wolcott Canned Foods	27-1/2 mi.	East
C. W. Stuart & Co. Newark Nurserymen	19 mi.	Southeast
Francis Leggett Co., Sodus Canned Foods	12-1/2 mi.	East
The Waterman Food Products Co. Food Processing	3-4 miles	South
Ontario Kraut Corp. 7 Railroad Ave. Food Processing	3-4 miles	South SW
Victor Preserving Co. Food Processing	3-4 miles	South
Ontario Cold Storage Food Processing	3-4 miles	South SW
Waterman Fruit Products Co. Food Processing	3-4 miles	South SW
Ontario Food Products Food Processing	3-4 miles	South SW
Lyndan Products Co. Food Processing	3-4 miles	South SW

September 29, 1981

Docket No. 50-244
LS05-81-09-074

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPIC II-1.C, POTENTIAL HAZARDS DUE TO NEARBY TRANSPORTATION, INSTITUTIONAL, INDUSTRIAL AND MILITARY FACILITIES - R. E. GINNA

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This completes our evaluation of Topic II-1.C.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

SE04
5
1/1
DSU USE (07)
ADD:
Alan Wong

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

OFFICE	SEPB:DL	~~SEPB:DL~~	SEPB:DL	ORB#5:DL:PM	ORB#5:DL:C	AD:SA:DL	
SURNAME	GCwalina:dk		WRussell	RSnaider	DCrutchfield	GLainas	
DATE	9/2/81		9/21/81	9/24/81	9/25/81	9/28/81	

Mr. John E. Maier

cc

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Bureau of Nuclear
Operations
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

U. S. Environmental Protection Agency
Region II Office
ATTN: Regional Radiation Representative
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

R. E. GINNA
SYSTEMATIC EVALUATION PROGRAM
II-1.C, POTENTIAL HAZARDS DUE TO
NEARBY TRANSPORTATION INSTITUTIONAL INDUSTRIAL AND
MILITARY FACILITIES

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for hazards originating at nearby facilities.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic III-4.D, "Site Proximity Missiles" reviews the extent to which the facility is protected against missiles originating from offsite facilities.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity."

V. EVALUATION

There is little industrial activity in the vicinity of the Ginna plant. Wayne County, where Ginna is located, is primarily a rural area. Typical industries for Wayne County are shown in Table 2.5-1 of the FSAR, reproduced here as Table 1. The nearest concentration of industrial activity is located in the town of Webster, about 6 miles from the site, and consists primarily of light manufacturing (Xerox copiers). No industrial development is expected to occur in the vicinity of the Ginna site.

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The nearest railroad to the plant is the Ontario Midland railroad about 3 1/2 miles to the south. Comparing this distance with the guidance provided in Regulatory Guide 1.91, it is apparent that potential railroad accidents involving hazardous materials are not considered to be a credible risk to the safe operation of the plant.

The nearest large pipelines to the plant are a 12" gas line located about six miles southwest of the plant, and a 16" gas line located about 10 miles south of the plant. These pipelines are far enough removed to assure that pipeline accidents will not affect the safety of the nuclear plant. Figure 1 shows a portion of the residential gas lines serving homes along Lake Road, as well as the house heating boiler at the Ginna plant itself. There are no gas or oil production fields, underground storage facilities, or refineries in the vicinity of the plant.

The potential effect of the gas line service to the Ginna house heating boiler was discussed during the Ginna Fire Protection review. This 4-inch gas line comes into the plant underground until it penetrates the ground surface at the east end of the screenhouse. This routing ensures separation from all other safety-related structures and systems. At this point, a metering station and a gas shutoff valve are located (the gas meter was relocated as a result of the Fire Protection review, item 3.1.13). The gas line is buried underground again after the gas meter regulator station, and enters the building through the basement wall under the house heating boiler area. The gas pipe is of welded steel construction up to the boiler. There is continuous ventilation of the areas that the gas line passes through within the building. The gas line service to the boiler and the boiler controls were reviewed and compared to NFPA-85, as requested in the staff's Fire Protection SER, dated February 14, 1979 (item

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There are no large commercial harbors along the southern shore of Lake Ontario near the plant. Some freight is shipped through Rochester harbor about 20 miles to the west. Major shipping lanes in the lake are located well off-shore, at least 23 miles or more, from the plant.[7] The possibility of damage to the service water intake structure was also considered. Section III-B.27 of RG&E's "Technical Supplement Accompanying Application for a Full-Term Operating License," August 1972 discusses the design of the intake system. As noted in this report, the intake system is completely submerged below the surface of the lake. A ten-foot reinforced concrete lined tunnel, driven through bedrock, extends 3100 feet northerly from the shoreline. The tunnel rises vertically and connects to a reinforced concrete inlet section. The occurrence of historical low water level will result in a depth of water of 30 feet at the inlet and with 15 feet of cover over the inlet structure. This is sufficient to prevent damage from any boating which might pass in the vicinity of the structure. Further, plugging of inlet water flow by a single large piece of material is prevented by the design of the inlet structure, in that water enters on a full 360° circle.[13] Another design feature

at Ginna to ensure continued availability of essential service water is that service water intake can be directly drawn from the discharge canal, which is located on the plant site, protected from any potential lake boating. Thus, lake navigation is not considered to be a hazard to the plant.

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The Williamson Flying Club Airport has one paved runway. This runway, designated 10-28 and thus oriented in an almost east-west direction, is 3377 feet long and 40 feet wide. The main runway is equipped with low intensity runway lights. The airport has instrument approach capability to runway 28 from the Rochester VORTAC. Figure 2 shows the instrument flight path. There is no control tower at this airport. The airport is used for general aviation activities such as business and pleasure flying, and for agricultural spraying operations. There are currently about 5,000 operations per year at the facility, and about 30 based aircraft, including part-time based crop dusters. The great majority of the aircraft are single-engine propeller airplanes which typically weigh on the order of 1500 to 3600 pounds.[8]

The small number of operations at this airport is substantially fewer than the criteria given in Section III.3 of SRP 3.5.1.6 and therefore is not considered a potential hazard.

Monroe County Airport, in Rochester, New York, located about 25 miles southwest of the plant, is the nearest airport with scheduled commercial air service. Low altitude federal airways V2 and V2N pass about 10 miles south and 2 1/2 miles southwest of the plant, respectively. The low altitude federal airways, V2 and V2N, serve about 10 flights per day. Almost all flights use V2, with V2N being used only occasionally. At most, 10% of airline traffic would use V2N. The width of these airways are eight miles.[9] We have reviewed the probability for an airline crash from these airways in accordance with the method given in SRP 3.5.1.6 Section III-2. The calculated probabilities are 5.1×10^{-8} for airway V2 and 1.4×10^{-8} for airway V2N. Since both airways probabilities are less than the 1×10^{-7} acceptance criteria, we conclude that the probability of a commercial air traffic crash at Ginna is acceptable.

Air Force Restricted Area R-5203 is located about eight miles north of the plant site. Whenever flight activity is conducted by the Air Force within R-5203, radar surveillance is maintained by the 21st NORAD Region, the 108th Tactical Control Group, or possibly the Cleveland Air Route Traffic Control Center. Pilots rely upon on-board navigational equipment to maintain their presence within the specified limits of the restricted

area. Pilots can also be advised if their aircrafts stray beyond their limits by the radar surveillance unit covering the area at the time. The restricted area is used daily for military flight training which includes high-speed interceptor training maneuvers, operational flight checks, and air-to-air refueling. The current altitude ranges from 2,000 to 50,000 feet above the surface.[5] A portion of the Detroit Sectional Aeronautical Chart, reproduced as Figure 3, shows the airports, air routes, and training space described above. There is also a slow-speed low altitude military training route (SR-826) which passes about 6 miles west of the plant. Acceptance criterion II.2 of SRP 3.5.1.6 states that, for military air space, a minium distance of five miles is adequate for low level training routes, except those associated with unusual activities, such as practice bombing. Air Force Restricted Area R-5203 is about eight miles from at its closest boundary, and no unusual activities such as practice bombing take place. The slow-speed low altitude military training route SR-826 is about 6 miles from the plant. Therefore, this criterion is met.

VI. CONCLUSION

Since current regulatory criteria are met with regard to SEP Topic II-1.C, it can be concluded that this topic is complete for the R. E. Ginna site. No additional review for this topic is required during the SEP integrated assessment.

References

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.5.
2. Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Sections 2.2.1, 2.2.2, 2.2.3, and 3.5.1.6, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Sterling Power Project Nuclear Unit No. 1 Preliminary Safety Analysis Report Addendum, Rochester Gas and Electric, Volume 1, Sections 2.1 and 2.2.
6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, Rev. 1, February 1978.
7. Conversation with Chief, U.S. Coast Guard Station, Rochester, New York, 4/8/81.
8. Conversation with Vern Tyrrell, manager of the Williamson Flying Club Airport, 4/7/81.
9. Conversation with FAA controller, Monroe County Airport, 4/8/81.
10. Fire Protection SER, Dennis L. Ziemann to Leon D. White, Jr., February 14, 1979.
11. Fire Protection SER, Supplement No. 2, Dennis M. Crutchfield to John E. Maier, February 6, 1981.

12. Letter, Dennis M. Crutchfield, NRC, to John E. Maier, RG&E, SEP Topics II-3.A, II-3.B, II-3.B.1, II-3.C, dated April 10, 1981.

13. Rochester Gas and Electric Corporation, "Technical Supplement Accompanying Application for a Full Term Operating License," August 1972.

Table 1

FSAR TABLE 2.5-1

TYPICAL INDUSTRIES IN WAYNE COUNTY

Company and Product	Distance from Site	Direction from Site
National Distillers & Chemical Corp. (Kordite Div.), Macedon Polyethelene Products	14-1/2 mi.	South
Duffy-Mott Co., Inc. Williamson Baby Foods	8-1/2 mi.	Southeast
Garlock, Inc. Palmyra Mechanical Packings	15 mi.	Southeast
Bloomer Bros. Co. Newark, Folding Paper Boxes	19 mi.	Southeast
Jackson Perkins Co. Newark Nurserymen	19 mi.	Southeast
Sarah Coventry, Inc. Newark Direct-mail sales of costume jewelry	19 mi.	Southeast
National Biscuit Co. (Dromedary Co. Div.) Lyons, Cake mixes, dates and peels	19 mi.	Southeast
General Electric Co., Clyde Electronic Equipment	27-1/2 mi.	Southeast
Comstock Foods Inc., Red Creek Canned Foods	31 mi.	East
Kenmore Machine Products, Inc. Lyons Refrigerant Products	22 mi.	Southeast
Olney & Carpenter, Inc. Wolcott Canned Foods	27-1/2 mi.	East
C. W. Stuart & Co. Newark Nurserymen	19 mi.	Southeast
Francis Leggett Co., Sodus Canned Foods	12-1/2 mi.	East
The Waterman Food Products Co. Food Processing	3-4 miles	South
Ontario Kraut Corp. 7 Railroad Ave. Food Processing	3-4 miles	South SW
Victor Preserving Co. Food Processing	3-4 miles	South
Ontario Cold Storage Food Processing	3-4 miles	South SW
Waterman Fruit Products Co. Food Processing	3-4 miles	South SW
Ontario Food Products Food Processing	3-4 miles	South SW
Lyndan Products Co. Food Processing	3-4 miles	South SW

FIGURE 1

Small Gas Lines in the Vicinity of Ginna

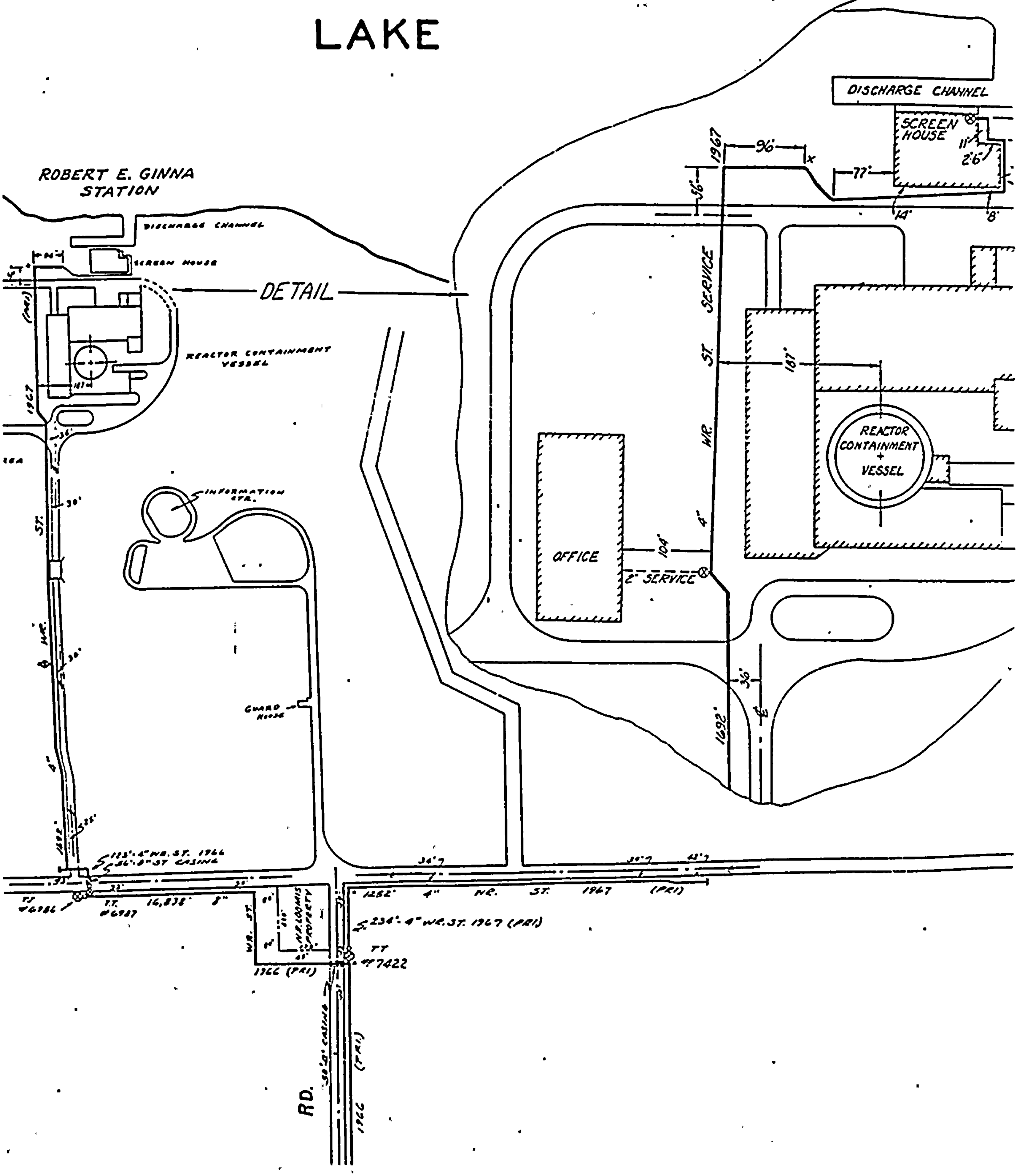


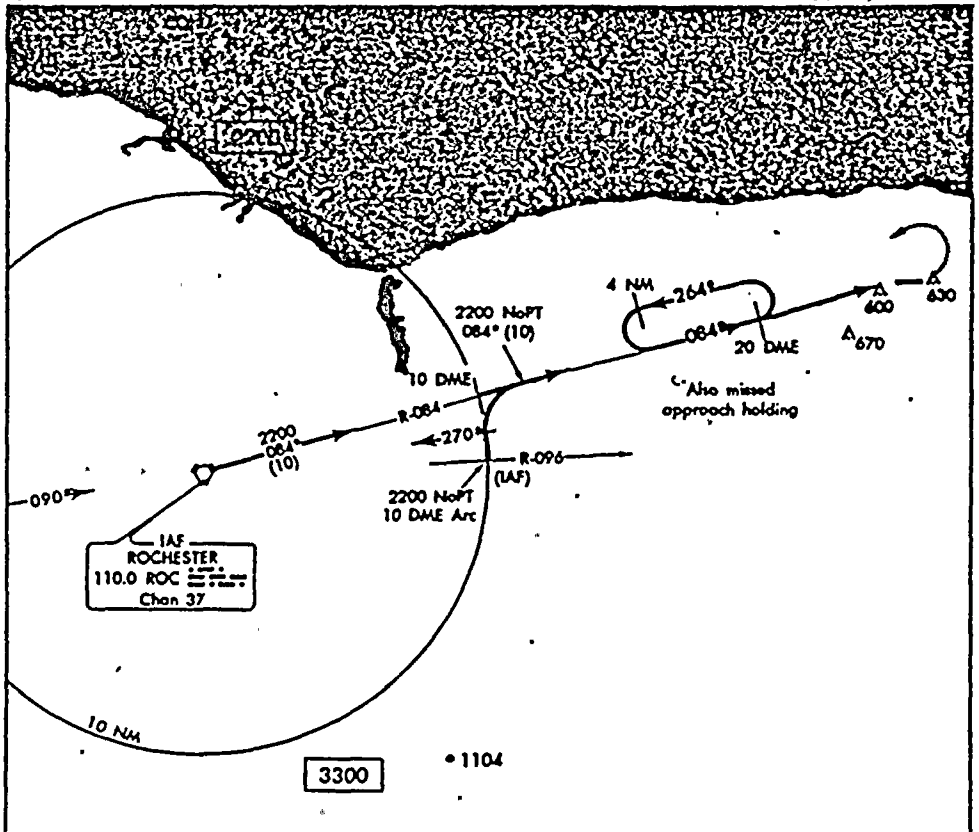
FIGURE 2

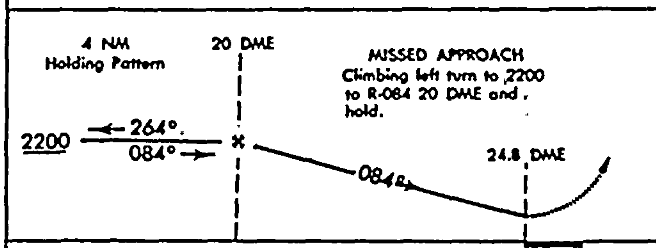
Instrument Landing Path to WFC Airport

Orig
VOR/DME RWY 10

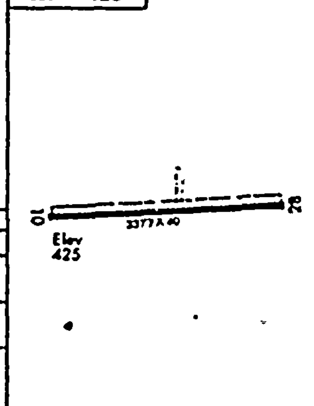
AL-6214 (FAA)

WILLIAMSON-SODUS
SODUS, NEW YORK





ELEV 425



CATEGORY	A	B	C	D
S-10	1300-1 875 (900-1)	1300-1¼ 875 (900-1¼)	1300-1½ 875 (900-1½)	1300-1¾ 875 (900-1¾)
CIRCLING	1300-1 875 (900-1)	1300-1¼ 875 (900-1¼)	1300-1½ 875 (900-1½)	1300-2 875 (900-2)

Use Rochester, New York altimeter setting.

▽

△ NA

LIRL Rwy 10-28

Knots	60	90	120	150	180
Min:Sec					

VOR/DME RWY 10

43°14'N-77°07'W

SODUS, NEW YORK
WILLIAMSON-SODUS

FIGURE 3

Detroit Sectional Aeronautical Chart - Rochester Vicinity