Safety Evaluation Report

NUREG-0091

U. S. Nuclear Regulatory Commission

related to construction of

Office of Nuclear Reactor Regulation

Montague Nuclear Power Station, Units 1 and 2

Docket Nos. 50-496
50-497

Northeast Nuclear Energy Company, et. al

July 1976

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NUREG-0091
July 1976

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

IN THE MATTER OF

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MONTAGUE NUCLEAR POWER STATION UNITS 1 & 2

DOCKET NOS. 50-496 AND 50-497

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Northeast Nuclear Energy Company acting for itself and for the 29 utilities listed in Table 1.1, filed with the Nuclear Regulatory Commission (NRC or Commission) an application, docketed on July 12, 1974, for licenses to construct and operate the proposed Montague Nuclear Power Station, Units 1 and 2 (Montague 1 and 2 or facility). The above cited utilities have designated the Northeast Nuclear Energy Company as their agent with responsibility for the technical adequacy of the design, construction, and operation of Montague 1 and 2. The Northeast Nuclear Energy Company and the 29 other utilities are hereinafter referred to as the applicants. The facility will be located in Franklin County, Massachusetts approximately 3.5 miles east-southeast of the town of Greenfield, Massachusetts.

Montague 1 and 2 will utilize the General Electric boiling water reactor (BWR-6) nuclear steam supply system with the MARK III type containment. The NRC review of the Montague 1 and 2 application was performed in accordance with its custom design review plan. However, with respect to the nuclear steam supply systems for Montague 1 and 2, since the applicants have stated that they are the same as that utilized for the GESSAR-238 nuclear plant and described in the General Electric Standard Safety Analysis Report (GESSAR) and have incorporated this description in the Preliminary Safety Analysis Report for Montague 1 and 2, we have relied on the evaluation we previously performed for the GESSAR application (Docket No. STN-50-447).

Since the time the application was docketed, i.e., on July 12, 1974, the applicants have announced two delays, a one-year delay and a four-year delay, in the start of construction for the Montague Nuclear Power Station, Units 1 and 2. The applicants indicated that this five-year delay was the result of a reappraisal of their capital construction program. The present estimated commercial operation dates for Montague 1 and 2 are April 1, 1986 and January 1, 1988, respectively.

We have completed our review of this application in all radiological safety areas to the extent possible at this time. Certain safety matters remain outstanding. These matters are described in Section 1.8 of this report. We plan to complete our review of these outstanding matters, and of any new significant safety considerations that develop in the interim, during our update review. We expect to initiate our update review about a year prior to when a decision from an Atomic Safety and Licensing Board, on issuance of construction permits for the facility, will be needed to permit the applicants to meet their start of construction date. The applicants have estimated this date to be in 1979 or 1980. Our update review could therefore start as early as 1978. Since our update review will be completed more than two years from now we

TABLE 1.1

LIST OF MONTAGUE NUCLEAR POWER STATION, UNITS 1 & 2 OWNERS

1. The Connecticut Light and Power Company
2. The Hartford Electric Light Company
3. Western Massachusetts Electric Company
4. Northeast Nuclear Energy Company*
5. New England Power Company
6. Central Maine Power Company
7. New Bedford Gas and Edison Light Company
8. Montaup Electric Company
9. Central Vermont Public Service Corporation
10. Fitchburg Gas and Electric Light Company
11. Burlington (Vermont) Electric Department
12. Town of Reading (Massachusetts) Municipal Light Department
13. City of Chicopee (Massachusetts) Municipal Lighting Plant
14. City of Holyoke (Massachusetts) Gas and Electric Department
15. Peabody (Massachusetts) Electric Department
16. City of Westfield (Massachusetts) Gas and Electric Light Department
17. Town of Shrewsbury (Massachusetts) Municipal Light Department
18. Town of Wakefield (Massachusetts) Municipal Light Department
19. Town of South Hadley (Massachusetts) Electric Light Department
20. Town of Hudson (Massachusetts) Light and Power Department
21. Marblehead (Massachusetts) Municipal Light Department
22. North Attleborough (Massachusetts) Municipal Light Department
23. Holden (Massachusetts) Municipal Light Department
24. Town of Littleton (Massachusetts) Electric Light and Water Department
25. Town of West Boylston (Massachusetts) Municipal Lighting Plant
26. Ashburham (Massachusetts) Municipal Light Plant
27. Town of Boylston (Massachusetts) Municipal Light Department
28. Paxton (Massachusetts) Municipal Light Department
29. Sterling (Massachusetts) Municipal Electric Light Department
30. Templeton (Massachusetts) Municipal Light Plant

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*In addition to being a co-owner of the proposed Montague Station, the Northeast Nuclear Energy Company is responsible for the design, construction and operation functions.

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would expect th' all the outstanding safety matters discussed in this report will have been resolved. The report we plan to issue upon completion of our update review should, therefore, describe the resolution of all the outstanding matters discussed in this Safety Evaluation Report.

The information provided for our review consisted of the Preliminary Safety Analysis Report, including Supplements 1 through 7, contained in Amendments 1 through 14 to the application. Copies of this report and its amendments are available for public inspection at the U. S. Nuclear Regulatory Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Carnegie Library, Avenue A, Turners Falls, Massachusetts 01376.

This report summarizes the results of our technical evaluation of the proposed Montague Nuclear Power Station, Units 1 and 2 performed by the Commission's staff, and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the facility. Additional details as to the scope and bases used by the Commission's staff to evaluate the radiological safety aspects of proposed nuclear power plants are provided in the Nuclear Regulatory Commission's Standard Review Plan, NUREG-75/087. The Standard Review Plan is the result of many years of experience by the Commission's staff in establishing and promulgating standards to enhance the safety of nuclear facilities and in assessing Safety Analysis Reports. Aspects of the environmental impact considered in the review of the application in accordance with 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection" of the Commission's regulations implementing the requirements of the National Environmental Policy Act of 1969 are discussed in the Commission's Final Environmental Statement which is presently scheduled for issuance in July 1976.

The review and evaluation of the proposed design of the facility reported herein is the first stage of a continuing review by the Commission's staff of the design, construction and operating features of Montague 1 and 2. Construction will be accomplished under the surveillance of the Commission's Office of Inspection and Enforcement. Prior to a decision for issuance of an operating license, we will review the final design to determine that all of the Commission's safety requirements have been met. The facility may then be operated only in accordance with the terms of the operating license and the Commission's regulations under the continued surveillance of the Commission's Office of Inspection and Enforcement.

Subject to favorable resolution of the outstanding issues discussed herein and summarized in Section 1.8 of this report, we will be able to conclude that the proposed Montague Nuclear Power Station, Units 1 and 2 can be constructed and operated without endangering the health and safety of the public. Our detailed conclusions are presented in Section 21.0 of this report.

1.2 General Plant Description

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Units 1 and 2 of the Montague Nuclear Power Station will each have a nuclear steam supply system which includes a boiling water reactor of the BWR-6 class with a rated

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power level of 3579 megawatts thermal. Each system will have 20 jet pumps supplied by two recirculating water lines, four main steamlines, and two feedwater lines. Fuel rods for the reactor will contain slightly enriched uranium oxide in the form of sintered ceramic pellets. Some of the fuel rods will contain gadolinium oxide in a mixture with uranium oxide, also in the form of sintered ceramic pellets. The gadolinium will serve as a "burnable poison", designed to flatten the power distribution and reduce the core reactivity variation throughout the core lifetime.

The fuel pellets will be enclosed in Zircaloy-2 tubes (cladding) which will be evacuated, backfilled with helium and sealed by welding Zircaloy-2 end plugs in each end. A Zircaloy-4 fuel channel will enclose a bundle of 63 fuel rods in an 8 x 8 array (one fuel rod position will contain a water filled rod). Water flowing through the core will serve as both a neutron moderator and as a coolant. Movement of water and the two phase water-steam mixture through the core will be accomplished by the driving force from the 20 jet pumps (10 per recirculation line), the two recirculation pumps, and from convective forces. Steam from the boiling process in the reactor core will be separated from the recirculating water, dried in the upper region of the reactor vessel, then directed through the four main steamlines to the turbine-generator system where its energy will be converted into electricity. The steam will then be exhausted to a condenser located beneath the turbine where the condensate will be collected and ultimately returned through a cleanup system for recycling through the reactor vessel and core. The cooling water for the condenser will recirculate through natural draft cooling towers, with makeup water taken from the Connecticut River. Blowdown from the cooling towers will be discharged to the Connecticut River.

The reactor coolant pressure boundary will include the reactor vessel, the recirculation lines, main steamlines, feedwater lines, and branch lines to their outermost isolation valves. Enclosing the reactor coolant pressure boundary (except for certain penetration lines) will be a reinforced concrete cylindrical structure (drywell). Enclosing this drywell will be a free-standing steel cylindrical structure (primary containment). Enclosing the primary containment will be a reinforced concrete cylindrical structure (shield building or secondary containment). The function of the drywell is to force most of the steam released during a postulated reactor coolant pressure boundary break, through the suppression pool located at the bottom of the primary containment thus condensing the steam and limiting the pressure buildup below the primary containment's design pressure of 15 pounds per square inch gage. Piping restraints will be installed within the containment to limit the movement of piping during postulated post-rupture movement (pipe whip) so that safety related components and systems are appropriately protected. A hydrogen control system will be provided to maintain the concentration of hydrogen within an acceptable range for the duration of all postulated accidents. Short term hydrogen buildup in the drywell will be controlled by a mixing system which will mix the relatively small drywell volume with the much larger primary containment volume. Long term hydrogen buildup will be controlled by hydrogen-oxygen thermal recombiners.

Isolation of the primary containment will occur whenever there exists a potential for the uncontrolled release of radioactivity.

The reactor protection system will provide the means to protect against conditions that may cause fuel failures or a breaching of the nuclear system process barrier. This system will initiate a reactor scram following an abnormal operational transient or pressure pulse, or following a gross failure of fuel or of the nuclear system process barrier. The reactor protection system will be a highly reliable system designed to meet the requirements of the Institute of Electrical and Electronics Engineers Standard-279.

The primary containment will house the reactor and its pressure suppression system. The auxiliary building will house the engineered safety features' auxiliary equipment. The fuel building will house the fuel storage and shipping areas. Operation of the standby gas processing systems will produce a negative internal pressure such that the atmosphere within the shield, auxiliary, and fuel buildings will be filtered and discharged via these systems to the plant vent. Other structures such as the turbine building, diesel-generator building, control building, radwaste building and the standby cooling towers are described in the appropriate sections of this report.

Engineered safety features will provide the capability to contain fission products assumed to be released during a hypothesized design basis accident to restrict radioactivity releases to acceptable levels, provide for heat removal for emergency core cooling, and condense steam within the containment. Details on these engineered safety features are presented in Section 6.0 of this report.

The radioactive waste management systems will collect, treat, and dispose of radioactive waste in a controlled and safe manner such that the discharges from the facility will be as low as practicable in accordance with the requirements of 10 CFR Part 50 and well within the limits of 10 CFR Part 20. Gaseous waste disposal systems will provide for collection, monitoring, purification, and holdup of noncondensible radioactive gases or suspended radioactive materials. During our review of the Final Safety Analysis Report, we will establish technical specifications that require gaseous effluents to be maintained at acceptable concentration levels before release from the plant's exhaust, which is located on top of the shield building. Liquid radioactive wastes will be collected, monitored, and processed to assure that releases to the Connecticut River will be within allowable limits. Solid wastes will be collected, drummed, and shipped to Commission approved offsite burial grounds.

1.3 Interaction Between Units 1 and 2

The Montague 1 and 2 is a two unit facility which has been designed for nearly complete separation of critical safety-related equipment and systems. Units 1 and 2 will share two standby cooling towers. Each standby cooling tower will consist of four independent cells with one fan per cell, all supported above a common water storage basin. The two standby cooling towers will be sized such that each is capable

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of supplying the required cooling capacity for a design basis accident in one unit and a simultaneous safe shutdown and cooldown of the other unit. Connections will be provided such that the Unit 1 standby cooling tower water storage basin can simultaneously supply the standby service water pump suction wells for both Units 1 and 2. The same situation will exist for the Unit 2 standby cooling tower water storage basin. The four fans in each standby cooling tower will be electrically arranged in groups of two. There are three standby diesel generators for each unit of the facility one of which is used solely for the high pressure core spray.

One group of fans (Division 1) in the Unit 1 standby cooling tower will be supplied from one of the Unit 1 standby diesel-generators (Division 1). Similarly, one group of fans (Division 2) in the Unit 2 standby cooling tower will be supplied from the other Unit 1 standby diesel generator (Division 2). The same logic will hold for the Unit 2 standby cooling tower fans. The Unit 1 and Unit 2 standby service water systems will have the capability for discharging to either the Unit 1 or Unit 2 standby cooling tower.

The radwaste system for treatment of liquid and solid radioactive wastes will be shared. Descriptions of these systems are provided in Section 11 of this report.

The operating organization for the proposed facility will provide for common management, service, and technical functions.

Based on our review of the interactions between the shared systems of Units 1 and 2, we conclude that they will not compromise the safety of either unit and are, therefore, acceptable.

1.4 Comparison With Similar Facilities

Many of the design features of the proposed facility are similar to those of BWR plants previously reviewed and approved and now under construction. To the extent feasible and appropriate, we have made use of our previous evaluations during our review of those features which are substantially the same as those considered for the earlier plants. Where this has been done, the appropriate sections of this report will include the identification of the other facilities involved. Our Safety Evaluation Reports for these other facilities are published and are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

A listing of the principal parameters and features of Montague 1 and 2 is presented in Table 1.2. The table also provides the comparable features for the River Bend (Docket Nos. 50-458/459) and Perry (Docket Nos. 50-441/442) plants. These plants were chosen for comparison since they have similar BWR-6 reactor and Mark III containment designs.

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TABLE 1.2

PRINCIPAL PARAMETERS AND FEATURES: MONTAGUE 1&2 AND COMPARABLE PLANTS

Parameter or Feature	Montague 1&2	River Bend 1&2	Perry 1&2
Rated Thermal Power (megawatts thermal)	3579	2894	3579
Design Thermal Power (megawatts thermal)	3758	3039	3758
Net Electrical Output (megawatts thermal)	1220	934	1205
Steam Flow Rate (pounds per hour)	15,396,000	12,451,000	15,396,000
Core Coolant Flow Rate (pounds per hour)	105,000,000	84,500,000	105,000,000
Feedwater Temperature (degrees Fahrenheit)	420	420	420
Normal Steam Pressure (pounds per square inch atmosphere)	1040	1040	1040
Separator Design Inlet Quality (percent)	14.88	14.88	14.88
Number of Fuel Assemblies	732	592	732
Number of Movable Control Rods	177	145	177
Reactor Vessel Design Pressure (pounds per square inch gage)	1250	1250	1250
Reactor Vessel Design Temperature (degrees Fahrenheit)	575	575	575
Reactor Vessel Inside Diameter (inches)	238	218	238
Reactor Vessel Inside Height (inches)	850	838	850
Number of Recirculation Loops	2	2	2
Circulating Pump Flow Rate	35,400	32,300	35,400
Recirculation Loop Inside Diameter (inches)	22/24	20	22/24
Number of Jet Pumps	20	20	20
Number of Steamlines	4	4	4
Steamline Inside Diameter (inches)	26	24	26
Number of Core Spray Spargers*	2	2	2
High Pressure Core Spray** System (gallons per minute at pounds per square inch difference)	1465@1130 6000@200	1325@1130 5010@200	1465@1130 6000@200
Pump Motive Type	Motor (Separate D-G)	Motor (Separate D-G)	Motor (Separate D-G)
Low Pressure Core Spray** System (gallons per minute at pounds per inch difference)	6000@122	5010@119	6000@122
Low Pressure Coolant Injection System			
Number of Pumps	3	3	3
Flow Rate (gallons per minute at pounds per square inch difference)	7100@20	5050@20	7100@20

*Separate Core Spray for the High Pressure Core Spray System and the Low Pressure Core Spray System.

**The High Pressure Core Spray System serves as a redundant Low Pressure Core Spray System.

TABLE 1.2 (Cont'd)

Parameter or Feature	Montague 1&2	River Bend 1&2	Perry 1&2
Number of Automatic Depressurization Systems	1	1	1
Residual Heat Removal System			
Number of Pumps	2	2	2
Flow Rate/Pump (gallons per minutes per pumps)	7100	5050	7100
Duty (British thermal units per Heat Exchanger)	45,000,000	35,000,000	45,000,000
Average Power	56.0	56.0	56.0
Maximum Design Linear Power (kilowatts per foot)	13.4	13.4	13.4
Maximum Heat Flux (British thermal units per square foot)	354,100	354,100	354,100
Maximum UO_2 Temperature (degrees Fahrenheit)	3325	3325	3325
Minimum Critical Power Ratio	$\geq$1.21	$\geq$1.21	$\geq$1.21
Total Peaking Factor	2.22	2.22	2.22
Fuel Rod Array	8x8 (63 fuel rods)	8x8 (63 fuel rods)	8x8 (63 fuel rods)
Fuel Rod Diameter (inches)	0.493	0.493	0.493

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1.5 Identification of Agents and Contractors

The General Electric Company will furnish the nuclear steam supply system and turbine generators for Montague 1 and 2, including the first fuel loadings. For those items of the facility within its scope of work, General Electric is acting as its own procurement agent.

Stone & Webster Engineering Corporation is the architect engineer and constructor. In this capacity they will be performing all of the architectural and engineering work, which includes the preparation of engineering specifications and design of all systems and components not supplied by General Electric. Stone & Webster Engineering Corporation will also construct Montague 1 and 2. In addition, they assist in the preparation of all license documents.

Other consultants retained by the applicants to perform or verify design concepts for the facility are identified in the Preliminary Safety Analysis Report.

1.6 Summary of Principal Review Matters

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed Montague 1 and 2 facility performed by the Commission's staff. Our evaluation included a technical review of the information and data submitted by the applicants with emphasis on the following principal matters:

(1) We evaluated the population density and land use characteristics of the site environs and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology to determine that these characteristics had been adequately described and were given appropriate consideration in the plant design, and that the site characteristics were in accordance with the Commission's siting criteria (10 CFR Part 100) taking into consideration the design of the facility, including the engineered safety features that are provided.

(2) We evaluated the design, fabrication, construction, testing, and expected performance of the plant's structures, systems, and components important to safety to determine that they are in accordance with the Commission's General Design Criteria (GDC). Other appropriate codes and standards, the Commission's Quality Assurance Criteria, and other appropriate guides have been identified and found acceptable.

(3) We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents, including high energy pipe failures outside the containment, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of

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their occurrence would be well within the Commission's guidelines for site acceptability as given in 10 CFR Part 100.

(4) We evaluated the applicants' plans for the conduct of plant operations, including the organizational structure, the technical qualifications of operating and technical support personnel, the measures taken for industrial security, and the planning for actions to be taken in the unlikely event of an accident that might affect the general public. Our evaluation in this area was designed to determine that the applicants are technically qualified to operate the facility and has established effective organizations and plans for safe operation of the facility.

(5) We evaluated the design of the systems provided for the control of the radiological effluents from the plant to determine that these systems will be able to control the release of radioactive wastes within the limits of the Commission's regulations as specified in 10 CFR Part 20, and that the plant will be operated in such a manner as to reduce radioactive releases to levels that are as low as is reasonably achievable in accordance with the Commission's regulations as specified in 10 CFR Part 50.

(6) We are evaluating the financial data and information provided by the applicants as required by the Commission's regulations (Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50) to determine that the applicants are financially qualified to design and construct the proposed facility. We will report the results of our evaluation in a supplement to this Safety Evaluation Report prior to consideration of this matter in the public hearing.

During our review, a number of meetings (see Appendix [illegible] to this report) were held with representatives of the applicants and the applicants' contractors and consultants to discuss various technical matters related to the facility. We also visited the site to assess specific safety matters related to the Montague Nuclear Power Station. A number of changes to the facility design were proposed by the applicants to reduce the probability of accidents and to better mitigate the consequences in the event an accident does occur. We reviewed these design modifications and found them to be acceptable. Specific details are provided in amendments to the Preliminary Safety Analysis Report and in appropriate sections of this report.

1.7 Requirements for Future Technical Information

During our reviews of recent applications for construction permits for the BWR-6 class of boiling water reactors, including the application for the GESSAR-238 Standard Nuclear Island Design, we have identified certain development programs that are applicable to these license applications, including the Montague 1 and 2 application. A listing of these programs is provided below.

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(1) Fuel surveillance program. (Section 4.2.1)
(2) Instrumentation for vibration and loose parts detection. (Section 4.6)

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(3) Safety relief valve surveillance program. (Section 5.2.2)
(4) Verification of the Mark III pressure suppression design. (Section 6.2.1.5)
(5) Mark III suppression pool dynamics. (Section 6.2.1.6)
(6) Core spray distribution. (Section 6.3.1)
(7) Study of effects of relief valves blowdown during various operating conditions. (Section 6.2.1.6)

The above listed programs are aimed at verifying the design for the nuclear steam supply system for the BWR-6 class of boiling water reactors and the design for the Mark III containment, and for confirming the associated design margins.

Based on our review, we conclude that the applicants have identified and will have performed the development program necessary for the design and safe operation of Montague 1 and 2 on a timely schedule, and that, in the event that results of any of this work are not successful appropriate restrictions on operations can be imposed or proven alternate designs can be utilized to protect the health and safety of the public.

1.8 Outstanding Matters

As a result of the applicants' delay of the anticipated date for the start of construction to 1979 or 1980, we have identified certain matters for which our review and conclusions are to be deferred to a date closer to a decision on issuance of the construction permits for the facility. These items are:

(1) Evaluation of the applicants' financial qualification (Section 20.0).

(2) Inspection by the NRC's Office of Inspection and Enforcement for implementation of the applicants' Quality Assurance Program (Section 17.0).

Approximately one year prior to a decision on issuance of the construction permits for Montague 1 and 2, the NRC staff will initiate an update review of the financial qualifications of the applicants and any new or generic matters which have safety significance in the design of Montague 1 and 2. Item 1 above will be addressed in a supplement to this report prior to a decision for issuance of the construction permits. Item 2 above will be completed when the procurement of equipment covered by the applicants' Quality Assurance Program has been initiated.

We have identified certain outstanding matters for which our review is not yet complete or the applicants have not provided an acceptable commitment at this time. Based on our review of these matters there is reasonable assurance that resolution by design modification or by establishment of design criteria, will not have a significant impact on the design of safety-related systems, structures, and equipment already reviewed and found acceptable by the NRC staff. We will require resolution of these matters prior to a decision for issuance of Montague 1 and 2 construction permits. The items in this category are:

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(1) Evaluation of the Anticipated Transients Without Scram (ATWS) generic resolution for BWR's (Section 7.2.1).

(2) Emergency Plan interface with Commonwealth of Massachusetts agencies (Section 13.3).

(3) Evaluation of reactor vessel shield wall and reactor vessel supports (Section 6.2.1.3).

In addition to the above matters, we have identified certain outstanding items which were identified and discussed in our "GESSAR-238 Nuclear Island Standard Design" Safety Evaluation Report, NUREG-75/110, dated December 1975. These issues also apply to the Montague 1 and 2 nuclear steam supply system. We are continuing to review these outstanding items with the objective of obtaining satisfactory resolution for each item. The results of our review and the final resolution for each of these items will be documented in a supplement to the above cited safety evaluation report. We require that our review on all of these outstanding matters be completed and that acceptable resolutions for each item be documented prior to a decision on issuance of construction permits for Montague 1 and 2.

The following is a listing and the review status of these outstanding items as of the writing of this report:

(1) The need to either upgrade the design classification to seismic Category I for the cooling water piping to the recirculation pumps, or provide an analysis to show failure of the recirculation pump seals will not result in excessive leakage. This matter is still outstanding. (Section 3.2.1)

(2) Provide for testability of the automatic depressurization system under operating conditions. We expect this matter to be resolved by mid-1976. (Section 7.3.1.2)

(3) The staff has completed its review of the design of the turbine trip system and the associated logic, and has found them to be acceptable. However, the analysis and consequences of turbine trip events are still outstanding. (Section 7.2.2)

(4) The rod control and information system design has been submitted by General Electric and is currently under review. (Section 7.6.1)

(5) Resolution of the staff concerns regarding the use and application of austenitic stainless steel in the GESSAR-238 nuclear steam supply system is dependent upon further information to be supplied by General Electric. (Section 4.5)

(6) We will require that General Electric provide a detailed description, a schedule and a commitment to implement the fast scram test program. (Section 15.2.2)

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(7) We will require that the GESSAR-238 application be updated to reflect the current status of the General Electric resolution to the issue of single failure protection for the residual heat removal system during shutdown cooling operations. (Sections 5.4.5 and 7.3.1.3)

(8) We will require that General Electric provide diverse interlocks on the low pressure core spray and low pressure coolant injection systems of the GESSAR-238 nuclear steam supply system to prevent the injection valve from opening in the presence of unacceptably high reactor pressure. (Section 7.3.1.3)

(9) We have completed our review of the environmental and seismic qualification design criteria for the GESSAR-238 nuclear steam supply system and have found them to be acceptable. However, documentation of the results of our review is still outstanding. (Sections 3.10.2, 7.8 and 7.9)

(10) The seismic classification of the safety-relief valve piping to the suppression pool discharge is still outstanding. We will require the piping to meet the requirements specified in Regulatory Guide 1.26 - Quality Group Classifications and Standards for Water-Steam-and-Radioactive-Waste-Containing Components of Nuclear Power Plants, and General Design Criteria 1 through 5 or demonstrate that the consequences of a failure of this piping are acceptable. (Section 6.2.1.6)

(11) We will require the dynamic system analysis for the Safe Shutdown Earthquake loads coincident with a steam line break to assure loads on the reactor internals are acceptable be provided prior to the decision on issuance of Montague 1 and 2 construction permits. (Section 3.9.1.4)

(12) We will require the five topical reports discussed in Section 4.3.7 of this report be submitted, reviewed, and accepted prior to a decision on issuance of construction permits.

We require that all of the above matters be acceptably resolved prior to a decision on issuance of construction permits for Montague 1 and 2 and will report the results of our review in a supplement to this Safety Evaluation Report. The present schedule for resolution and documentation of the above twelve items is late 1976.

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2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The Montague site is located in the northern region of the town of Montague, Franklin County, Massachusetts which is comprised of five distinct communities, Turners Falls, Millers Falls, Lake Pleasant, Montague Center and Montague City. The reactor site is 1.2 miles south-southeast of the village of Turners Falls (population 5,168), 1.5 miles west-southwest of Millers Falls (population 1,186), and 3.5 miles east-southeast of Greenfield (population 14,642). Figure 2.1 (Preliminary Safety Analysis Report Figure 2.1.3-1) shows the reactor location with respect to the surrounding centers of population. The 1900-acre site is owned by the applicants who thereby possess the requisite right of control over the activities within the exclusion area.

Figure 2.2, taken from a U.S. Geological Survey Map of Montague area, indicates the site to be in a hilly terrain with elevations ranging between 300 and 560 feet above mean sea level.

The applicants have specified a minimum exclusion radius of 2,674 feet (816 meters) from each of the two units as shown in Figure 2.3. The applicants have specified a low population zone of 2.5 miles (4,022 meters). The nearest permanent residents are approximately 1290 meters northeast of the reactors. On the basis of a comparison of the site suitability information submitted by the applicants regarding population, road network, and land use factors within the proposed low population zone for the Montague 1 and 2 with similar characteristics of previously approved sites, we find that there are no factors which would preclude the development of adequate emergency measures to protect the public therein, provided prompt notification is made to those persons located in the near vicinity of the site boundary, e.g., in the towns of Millers Falls, Lake Pleasant, and Turners Falls (see Section 13.3). We find that this prompt notification, in the event of a serious accident with radiological consequences, is feasible and that the associated measures can be developed during the operating license stage of our review.

Figure 2.4 shows the cumulative population projected for 1980 and for 2020 around the Montague site. The 1970 population within the low population zone is about 4,476 using the 1970 census figures, which agrees with the applicants' figures. The nearest population center (as defined in 10 CFR Part 100) with a present population exceeding 25,000 is Northampton, Massachusetts, 15.5 miles south-southwest from the proposed nuclear plant site. The applicants have specified a low population zone of 2.5 miles. In the event that Greenfield, Massachusetts (current population 15,000) should reach 25,000 persons or more and thus become the population center as defined in 10 CFR Part 100, the closest boundary of this population center to the facility

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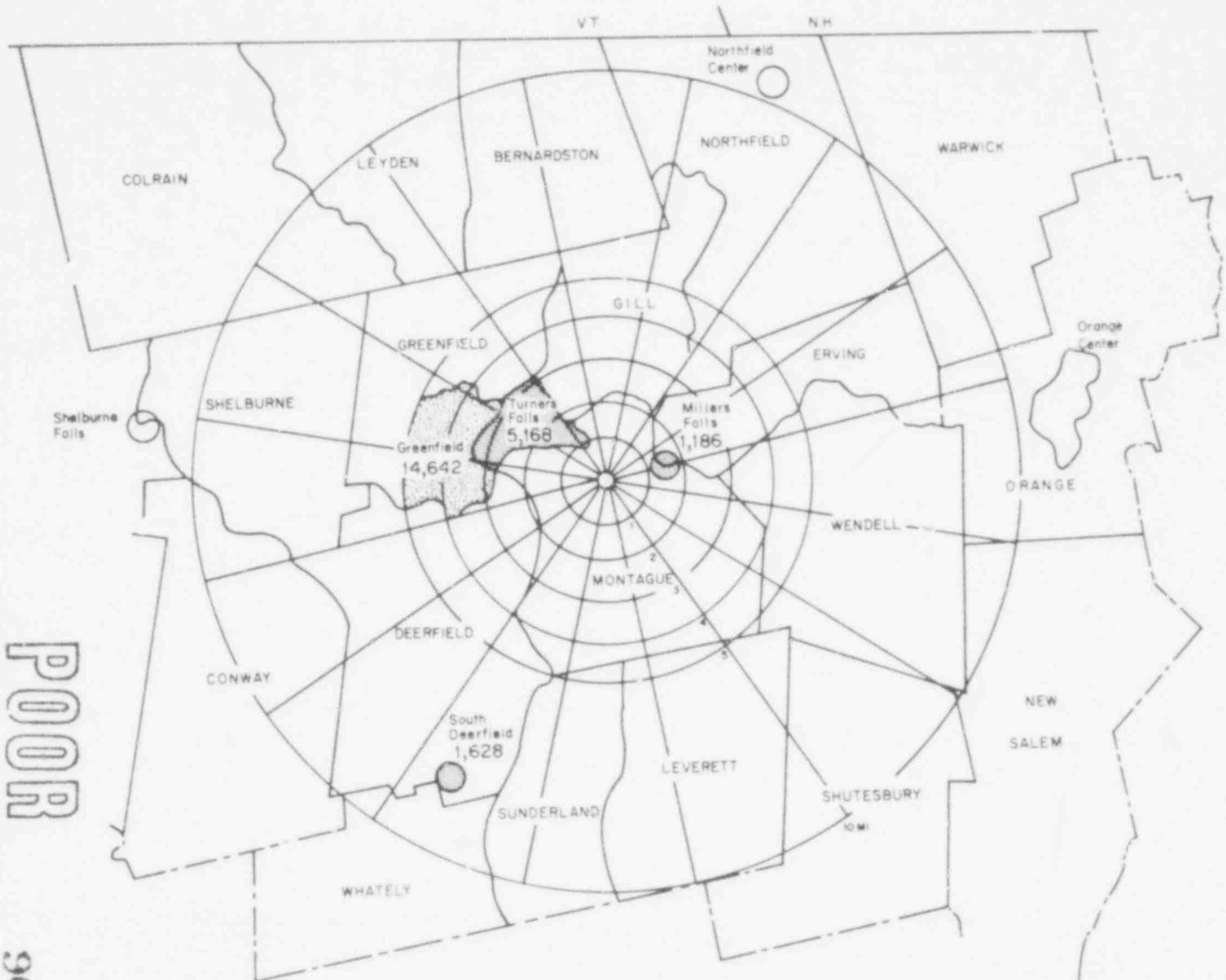

Figure 2.1 — CITIES AND TOWNS 1970 POPULATION — 10 MILES RADIUS

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0 2000 4000 6000 8000 10000
SCALE-FEET

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Figure 2.2 – STATION LOCATION

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Figure 2.3 – EXCLUSION AREA

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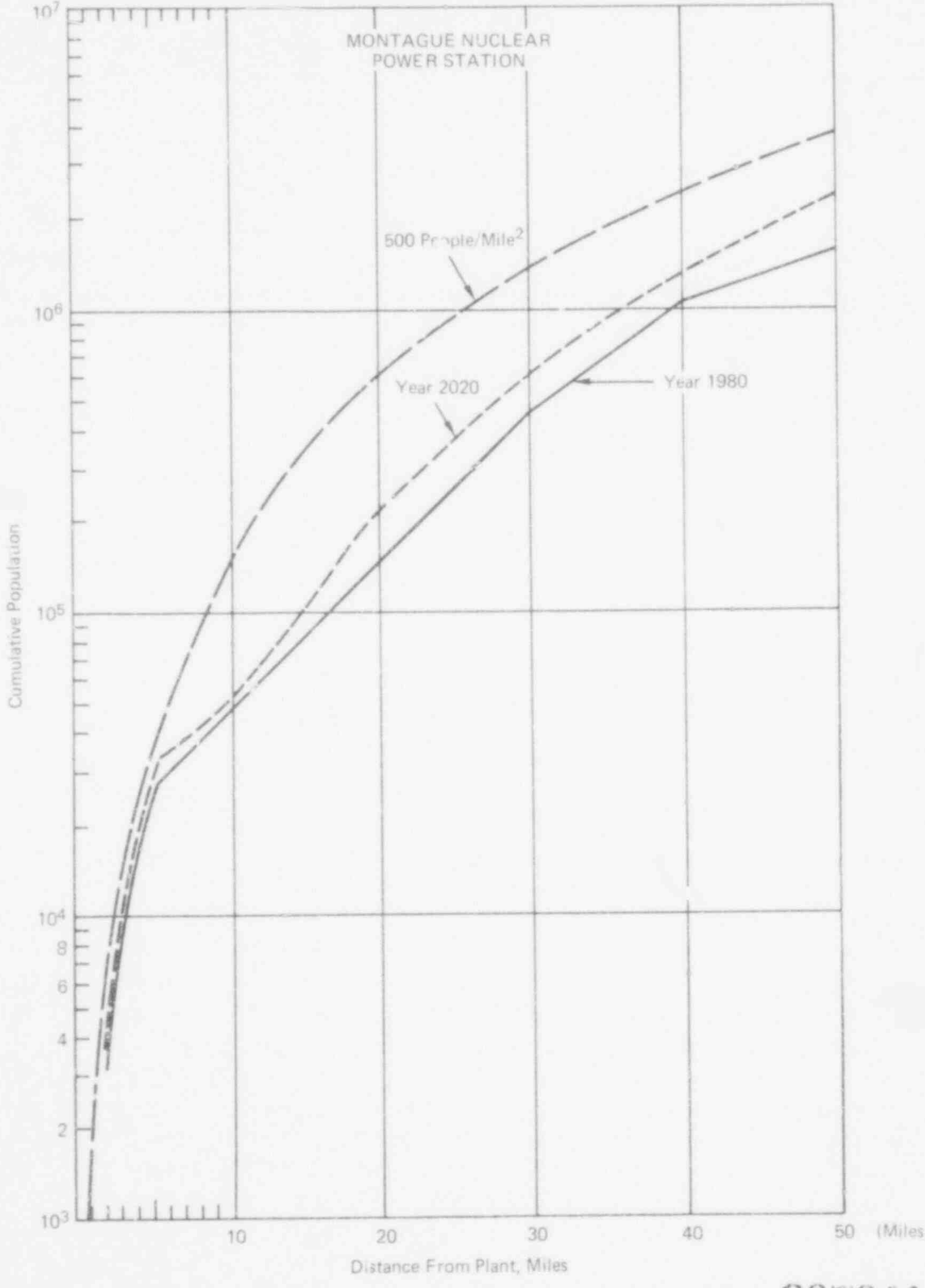

Figure 2.4 CUMULATIVE POPULATION DISTRIBUTION

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would be about 3.3 miles due to the location of the Rocky Mountain Park and the Connecticut River. This possible future population center distance of 3.3 miles would still be further than approximately one and one-third times the low population zone distance of 2.5 miles and would comply with the 10 CFR Part 100 guidelines.

The applicants predict that the population within a 50-mile radius will increase from a 1970 population of 1,741,221 to 2,419,337 in the year 2020, an increase of 39 percent. The population projections of the Bureau of Economic Analysis (BEA) for Economic Areas 4, 5, and 6 (Figure 2.5) are for a population increase of about 35 percent which is similar to the applicants' estimate.

The Connecticut River flows in an east to west direction about 1.6 miles northwest of the site. At the present time, the land bordering the Connecticut River is lightly populated with only two significant population groups at Greenfield and Turners Falls. There are three major employers within a five-mile radius of the site who employ a total of approximately 1500 people for three shifts. The Connecticut River is used for pleasure boating, water skiing, fishing, and swimming. There is no commercial water transportation on the Connecticut River in the vicinity of the Montague 1 and 2 due to the presence of dams (without locks) which preclude river traffic. Fishing activities on the Connecticut River near the site are comprised only of sport fishing. There is no commercial fishing within that stretch of river between Turners Falls and Holyoke Dam. The public facilities within the 2-1/2 mile low population zone include nine schools with a current enrollment of about 2,900 students.

Forests, private, public and semi-public, comprise about 80 percent of the land use in Franklin County. Residential and manufacturing use are two percent and one percent, respectively, with farm lands (cultivated and open land) totaling approximately 14 percent of the total number of acres in Franklin County. Recreational land and water use in the area of Montague 1 and 2 consists of boating, fishing, hiking, camping, swimming and snowmobile usage. There are a number of recreational areas within a 10-mile radius of the site. When considered on an annual basis, the average number of transients using recreational facilities in the site vicinity is not significant.

We conclude that the land and water uses have been adequately considered by the applicants and are not critical with respect to the operation of Montague 1 and 2. On the basis of the applicants' specified population center distance, minimum exclusion area, and low population zone, our analysis of the onsite meteorological data from which atmospheric dilution factors were calculated (Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents discussed in Section 15.0 of this report, we conclude that the proposed exclusion area, low population zone, and population center satisfy the requirements of 10 CFR Part 100 and are, therefore, acceptable.

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1 Bangor, Maine
2 Portland-South Portland, Maine
3 Burlington, Vt.
4 Boston, Mass.
5 Hartford, Conn.
6 Albany-Schenectady-Troy, N.Y.
7 Syracuse, N.Y.
8 Rochester, N.Y.
9 Buffalo, N.Y.

Figure 2.5 — ALBERS EQUAL AREA PROJECTION
Scale 1: 5,000,000

POOR ORIGINAL

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2.2 Nearby Industrial, Transportation, and Military Facilities

There are nine tank farms for gasoline, fuel oil, and propane located in the three to five-mile annular area from the plant site. These tank farms are at a sufficient distance to preclude a hazard to the proposed nuclear facility.

The Boston and Maine Railroad has a line that passes approximately 1.4 miles south and east of the site. The Central Vermont Railroad has a line that goes in a north-south direction on the eastern side of the Connecticut River about 1.4 miles from the proposed nuclear plant. Figure 2.3 shows the transportation routes within a three-mile radius of the site. A four-inch diameter low pressure (25 pounds per square inch-gage) gas pipeline passes 1.5 miles from the plant site. Based on the distances involved for these routes, we conclude that transportation accidents involving flammable gases shipped by these means will not present a hazard to the safety of the proposed nuclear plant.

The site proposed for Montague 1 and 2 is within a mile of the end of the runway at the Turners Falls Municipal Airport. This is a general aviation airport used primarily for instruction flights and for rental and sales operations. The runway is 3000 feet by 75 feet, paved with an asphalt surface, and with a 340/160 degrees orientation. Although the airport is capable of handling up to 30,000-pound aircraft, visual surveillance of operations by the applicants between December 1973 and October 1974, showed that no aircraft of over 12,500 pound gross weight utilized the airport.

The applicants have proposed to design all safety related structures of Montague 1 and 2 to withstand the impact of an aircraft weighing up to 15,000 pounds.

The present and projected traffic at the Turners Falls Airport is reported in Appendix G of the Preliminary Safety Analysis Report and indicates that 98 percent of the operations (landings or takeoffs) during a three month surveillance period were in a weight class of less than 5,000 pounds. As indicated above, no operations of aircraft weighing more than 12,500 pounds were observed between December 1973 and October 1974. In a recent letter from the Turners Falls Airport, we were informed that a Lockheed 18 "Lodestar" with a maximum weight load of 18,500 pounds is currently stationed at this airport.

Our criterion regarding the design of a nuclear reactor facility near an airport requires that the probability of an aircraft strike on the facility which could result in radiological consequences greater than 10 CFR Part 100 guidelines be less than one in ten million per year, and that the facility be hardened to protect safety related structures for aircraft which have a strike probability greater than one in ten million per year. Based on our analysis, we find that our criterion is satisfied for Montague 1 and 2 as long as there are not more than about one hundred operations per year of those aircraft weighing greater than 15,000 pounds. We will review this matter again prior to completion of the construction permit stage of our review to assure that the probability of one in ten million per year for aircraft weighing greater than 15,000 pounds is satisfied.

2.3 Meteorology

Information concerning the atmospheric diffusion characteristics of a proposed nuclear power plant site is required in order that a determination may be made that postulated accidental, as well as routine operational, releases of radioactive materials are within NRC guidelines. Further, regional and local climatological information, including extremes of climate and severe weather occurrences which may affect the safe design and siting of a nuclear plant at a proposed site, is required to insure that safety-related plant design and operating bases are within NRC guidelines. The meteorological characteristics of a proposed site are determined by the staff's evaluation of meteorological information in accordance with the procedures presented in Sections 2.3.1 through 2.3.5 of the USNRC Standard Review Plan, NUREG-75/087.

2.3.1 Regional Climatology

The applicants have provided a sufficient description of the regional meteorological conditions of importance to the safe design and siting of this plant.

The climate of northcentral Massachusetts is continental in character, with a large annual range in temperature and frequent, sometimes rapid, weather changes. Winters are long and cold while summers are short and warm. In summer, maritime tropical air masses with origins over the Gulf of Mexico or Caribbean Sea predominate over the region. During the other seasons, continental polar air masses from Canada are most frequent over the region. Temperatures of 90 degrees Fahrenheit or higher may be reached on about 10 days annually over this region while temperatures of zero degrees Fahrenheit or lower may be expected on seven days. On 150 days annually, temperature of 32 degrees Fahrenheit or lower may be expected. Precipitation is well distributed throughout the year, averaging about 50 inches annually. During the summer, precipitation occurs mainly as showers or thundershowers, while during the other seasons precipitation mainly occurs as rain or snow associated with large-scale migratory storm systems moving across the region. On an annual basis, relative humidity averages around 70 percent.

Severe weather occurrences at the Montague site are associated mainly with severe thunderstorms or with intense, large scale winter storm systems. Tropical storms or hurricanes infrequently affect the site. During the period 1955-1967, twenty-eight tornadoes were reported within the one degree latitude-longitude square containing the site, giving a mean annual frequency of 2.2 and a computed recurrence interval for a tornado at the plant site of 580 years.

There were 16 reports of hail three-quarters of an inch in diameter or greater in the one degree latitude-longitude square containing the site during the period 1955-1967, and 27 wind storms with wind speeds of 50 knots (58 miles/hour) or greater. Twenty-two tropical storms or hurricanes have passed within 50 miles of the Montague site during the period 1871 through 1972. The maximum fastest mile wind speed recorded at Worcester, Massachusetts, 40 miles southeast of the site, is 76 miles/hour. At Windsor Locks, Connecticut, 45 miles south of the site, the maximum fastest mile wind speed recorded is 70 miles/hour. Freezing precipitation may be expected to occur once or twice each winter in the vicinity of the site and one severe ice storm (accumulation of one inch or more) every five years. High air pollution potential (air stagnation) can be expected to occur on two days annually.

The design basis tornado used for the plant, with a maximum wind speed of 360 miles/hour consisting of a maximum rotational wind speed of 290 miles/hour and translational wind speed of 70 miles/hour, a maximum pressure drop of 3.0 pounds per square inch, and a maximum pressure drop rate of 2.0 pounds per square inch/second, conforms to the recommendations of Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, and is sufficient for the area of the country in which the plant is located. The design basis wind is a sustained (fastest mile) wind speed of 90 miles/hour at a height of 30 feet with a return period of 100 years. We conclude the design basis tornado and wind are acceptable for the Montague site.

2.3.2 Local Meteorology

The applicants have provided sufficient information for us to make an evaluation of the local meteorological conditions of importance to the safe design and siting of this plant.

Long-term weather records from Worcester, Massachusetts show that extreme maximum and minimum temperatures of 102 and -24 degrees Fahrenheit, respectively, have been recorded there. At Windsor Locks, Connecticut, the extreme maximum temperature of record is 102 degrees Fahrenheit and the extreme minimum temperature is -26 degrees Fahrenheit. The maximum 24 hour rainfall amount of record at Windsor Locks is 12.1 inches and at Worcester is 8.7 inches. The maximum 24 hour snowfalls recorded at Worcester and Windsor Locks are 24.0 inches and 19.0 inches, respectively. Weather records from Turners Falls, covering a shorter period of record, show that extreme maximum and minimum temperatures of 103 and -30 degrees Fahrenheit have occurred. The maximum 24 hour precipitation total recorded at Turners Falls is 4.61 inches, and the maximum 24 hour snowfall total is 15.0 inches. Thunderstorms may be expected to occur on about 25 days annually in the site vicinity and heavy fog (visibility one-quarter mile or less) on approximately 50 days. Wind data collected at the 33 foot level onsite during the period April 1974 through March 1975 show that the predominant wind flow direction over the site at this level is from the north-northeast, with a frequency of 11.5 percent. Winds from the east-southeast occurred least frequently (1.1 percent).

The 70 pounds/square foot weight of snow and ice on the ground used as the design basis for weights due to snow and ice on the roofs of safety-related structures for the extreme environmental condition is considered to be sufficient for this site.

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2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program has been compared with the recommendations and intent of Regulatory Guide 1.23-Onsite Meteorological Programs. The staff concludes that the meteorological measurements program has produced data which, in turn, have been summarized to provide sufficient meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents from the nuclear facility.

A 500 foot high meteorological tower, erected onsite 7,700 feet south of the proposed reactor site, became operational on September 1, 1973. Wind speed and direction, temperature, and dewpoint are measured at the 33 and 494-foot levels on the tower. In addition, wind speed and direction are measured at the 150 and 325-foot levels and visibility at the 15-foot level. In July 1975, additional wind speed and direction measuring equipment was installed at the 70-foot level. Vertical temperature difference measurements are made between the 33-foot and 494-foot levels, between the 33-foot and 325-foot levels and between the 33-foot and 150-foot levels. Solar radiation from the sky is measured at the 33-foot level. The system conforms to the recommendations of Regulatory Guide 1.23 - Onsite Meteorological Programs.

Although data collection began onsite in September 1973, vandalism at the meteorological tower in late February 1974 resulted in the loss of a considerable amount of data extending well into March 1974. The recovery rate for the initial year of data provided by the applicants therefore, was below the recommended 90 percent. Subsequently, the applicants have acquired a second full year period of data, collected from April 1974 through March 1975, with a recovery rate of over 90 percent.

The applicants have provided sets of joint frequency distributions of wind speed and direction by atmosphere stability class (based on vertical temperature difference) from the one-year period (April 1, 1974 to March 31, 1975) of onsite data with a recovery rate of 96 percent. One set of these distributions was based on the wind speed and direction at the 33-foot level and the vertical temperature difference between the 33 and 150-foot levels, and another on the wind speed and direction at the 150-foot level and the vertical temperature difference between the 33 and 150-foot levels. In addition, the applicants have also provided a magnetic tape containing the hourly values of the wind speed and direction at the 33 and 150-foot levels and the vertical temperature difference between the 33 and 150-foot levels for this one year period of onsite data record.

An evaluation of these data indicated that although the data collected at the 33-foot level are representative of atmospheric dispersion conditions at the site in its present condition, the data may not adequately represent conditions that would be expected to exist after tree removal and plant construction at the site are completed. To resolve this concern, we requested the applicants to provide wind profiles (vertical distributions of wind speed with height) using the wind speed data from all levels on the tower where wind data were being collected. The applicants complied

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with this request. After reevaluation of this information, we concluded that the wind data collected at the 150-foot level, after appropriate conservative adjustments were applied, would best represent post-construction atmospheric dispersion conditions. These adjustments were accomplished by assuming a conservative displacement height of 20 feet (one-half the height of the nearby trees) to represent the effects of the tree canopy, which in turn assumes that the 150-foot winds represent winds at the 130-foot level above ground after the trees have been removed. The wind speeds were then adjusted, using the information obtained from the wind profiles, to represent conditions at the 33-foot level. These adjusted data were used in our evaluation.

The applicants have installed an additional wind measuring system at the 70-foot level on the tower (30 feet above the level of the tree tops) to collect data to better define the wind flow conditions that might be expected once the trees are removed. Data collection began at this level in July 1975, and will be provided to the staff for evaluation upon completion of an annual cycle.

2.3.4 Short-term (Accident) Diffusion Estimates

Conservative assessment of post-accident atmospheric diffusion conditions have been made by us from the applicants' meteorological data and appropriate diffusion models. In the evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the low population zone distance) accidental releases from the plant buildings and vents, a ground-level release with a building wake factor, cA, of 1180 square meter was assumed. Based upon terrain conditions at this site and the results of actual dispersion measurements at similar sites, the diffusion model described in Regulatory Guide 1.3 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactor was modified to include credit for lateral plume meander under stable atmospheric conditions with light wind speeds. The dispersion model also considered the actual site boundary distance in each of the sixteen 22 1/2° wind direction sectors around the plant and the frequency of winds blowing into each sector.

The relative concentration (X/Q) for the 0-2 hour time period following an accidental release, equivalent to that expected to be exceeded no more than five percent of the time at the exclusion distance, is 8.4×10^{-4} seconds/cubic meter. This value occurred in the north-northeast direction from the plant at the site boundary distance of 346 meters. The relative concentrations at the outer boundary of the low population zone (4022 meters) for the various time periods following an accidental release to the atmosphere are:

0-8 hours	5.6×10^{-5} seconds/cubic meter
8-24 hours	3.8×10^{-5} seconds/cubic meter
1-4 days	1.7×10^{-5} seconds/cubic meter
4-30 days	5.1×10^{-6} seconds/cubic meter

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2.3.5 Long-Term (Routine) Diffusion Estimates

Reasonable estimates of average atmospheric diffusion conditions have been made by the staff from the applicants' meteorological data and appropriate diffusion models as described in Regulatory Guide 1.111-Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactor, which is out for public comment. The highest offsite annual average relative concentration of 5.4 x 10^{-5} seconds/cubic meter for vent releases occurs at the site boundary north-northeast of the proposed reactor complex.

2.3.6 Conclusions

The applicants have provided sufficient information concerning those meteorological conditions which are of importance to the safe design and siting of the facility. The design basis tornado parameters used for the plant conform to the provisions of Regulatory Guide 1.76 - Design Basis Tornado for Nuclear Power Plants. The applicants' onsite meteorological program conforms to the provisions of Regulatory Guide 1.23 - Onsite Meteorological Program and has produced data which adequately describe site atmospheric dispersion conditions and which was used by the staff to make both conservative and realistic estimates of atmospheric dispersion characteristics for accidental and routine gaseous releases, respectively, from the plant. The applicants are expected to continue meteorological data collection at the site during the construction phase. These additional data are expected to provide sufficient information, concerning the atmospheric dispersion condition at the site as affected by plant construction, to verify these estimates.

2.4 Hydrology

2.4.1 Hydrologic Description

The proposed site is located in Franklin County, Massachusetts, and is within 1.5 miles of the Connecticut River at its nearest point. Plant grade is to be 340 feet mean sea level datum. Primary plant structures are to be located on the Montague Plain at the foot of the Wills Hill (peak elevation 566 feet mean sea level). An open channel is to be constructed around a section of the plant to collect runoff water from the Wills Hill, and thus eliminate the possibility of flooding from this source.

The proposed site is located on the inside of a bend in the Connecticut River. The drainage area for the Connecticut River at the Montague City gaging station (approximately two river miles upstream of the proposed site intake structure) is 7,865 square miles. Two tributaries of the Connecticut River near the site are the Deerfield and Miller Rivers, two and ten miles, respectively, upstream of the proposed intake structure.

There are 15 dams on the Connecticut River upstream of the proposed intake structure. The upstream dams within 50 miles of the intake structure (river mile 117) are the

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Turners Falls Dam (river mile 122, 17,000 acre-feet of usable storage) and the Vernon Dam (river mile 142, 12,000 acre-feet of usable storage).

Lake Pleasant and Green Pond are located approximately one mile southeast of the site and have a surface area of 53 acres and 15.3 acres, respectively. Together they comprise one of the three public water supplies for the Town of Montague. According to the annual report of the Turners Falls Fire District for the year ending December 31, 1973, a total of 77,950,000 gallons of water was pumped from the lake and pond in 1973 for use as domestic water supply.

No municipality downstream of the proposed Montague Station discharge derives its water supply from the Connecticut River. The applicants' well survey indicates as many as 96 water wells within about two miles of the site. The primary use of ground water in the area is for domestic water supply, although some ground water is used for agriculture. The Montague Well Field, located 2.5 miles south of the site area, furnishes part of the municipal water supply for the towns of Turners Falls, Millers Falls, Lake Pleasant and Montague City.

2.4.2 Flood Potential

Several possible flood-producing sources were discussed by the applicants. They include a probable maximum flood on the Connecticut River, dam failures on the Connecticut River, and local probable maximum precipitation at the plant site area including Wills Hill.

The Connecticut River is subject to severe flooding due to the topography and climate of the basin. The applicants have estimated the probable maximum flood at the Montague City Gage (two miles upstream of the proposed intake structure) by using: (1) the standard project flood calculated by the Corps of Engineers for the Connecticut River Basin; (2) the probable maximum flood estimated for the Turners Falls Dam (522,000 cubic feet/second); and (3) the Deerfield River probable maximum flood (79,000 cubic feet/second). The resultant probable maximum flood peak discharge estimate at the Montague City gage was 601,000 cubic feet/second with a corresponding stage of 190 feet mean sea level.

We independently estimated the probable maximum flood on the Connecticut River at the intake structure using the conservative method and information in a report by Nunn, Snyder and Associates. A probable maximum flood peak discharge value of 800,000 cubic feet per second (stage of 209 feet mean sea level) was estimated. Since this conservative probable maximum flood estimate resulted in a flood level more than 100 feet below the elevation of any safety-related structures or equipment, we conclude that the probable maximum flood on the Connecticut River would not be a safety concern. The failure of one of the upstream dams coincident with a standard project flood on the Connecticut River was determined not to be a problem because of the large difference in elevation between the standard project flood level (about 160 feet mean sea level) and plant grade (340 feet mean sea level).

The applicants evaluated the effects of a local probable maximum precipitation on the plant site area including Wills Hill, and have proposed constructing an open channel to collect the runoff from Wills Hill and divert it away from safety-related facilities. The applicants' design bases and proposed design of the channel were independently evaluated by the staff. In Supplement No. 7, the applicants agreed to revise their design either by increasing the depth of the channel by one foot, or by adding a one-foot high levee on the downhill side of the channel. We conclude that this design is acceptable.

The applicants propose to protect safety-related structures and equipment from flooding due to precipitation as severe as the local probable maximum precipitation on the plant. The plant yard is to be graded away from all safety-related structures. The exterior access to safety-related buildings is to be a minimum of six inches above ground grade. The openings to buildings in the area between the containment buildings (where ponding may occur) are to be a minimum of 2.5 feet above plant grade. Further, to prevent ponding of water on safety-related structures, the roofs of the structures are to be sloped and will not have parapets.

2.4.3 Water Supply

The makeup water intake structure for the proposed Montague Station will be located on the Connecticut River at river mile 117. This makeup water will be pumped to the normal and essential service water cooling tower basins.

Water necessary to shut down the plant under normal or accident conditions is to be taken via pumps located in the standby cooling tower pump well structure. The water is to be pumped from two seismic Category I basins, each with a 14,100,000-gallon capacity. A seismic Category I cooling tower will be located above each of the basins. The basins and cooling towers will be located northwest of the containment buildings at the base of Wills Hill. We conclude that the volume of water available in the basins will be adequate to meet the recommendations of Regulatory Guide 1.27-Ultimate Heat Sink for Nuclear Power Plants.

2.4.4 Ground Water

The three major aquifers in the site area are the Lake Bed, Bedrock, and Montague Wells. The Lake Bed Aquifer forms the surface of the Montague Plain, and is made up of medium sand with up to 20% gravel and numerous layers of fine and silty sand. Ground water in this aquifer exists at depths between 40 and 70 feet. Ground water in the Bedrock Aquifer within the Wills Hill area is under artesian pressure. The nearest edge of the Montague Wells Aquifer is located 2.5 miles south of the proposed site, and consists of boulders, clay, coarse gravel, sand, and silt.

Ground water movement on the Montague Plain is away from the site to Green Pond and Lake Pleasant in the southeast and toward Montague Road in the southwest.

The applicants do not plan to use ground water at or near the site during plant construction or operation.

The applicants proposed that the design water level for ground water-induced hydrostatic loading for all safety-related structures be plant grade. We conclude that this basis is conservative.

The applicants analyzed the effects of a postulated accidental spill of radioactive wastes into the ground water. The analysis is not conservative for the following reasons:

(1) The effects on the most critical water supply were not analyzed.

(2) In estimating the ground water velocities the effective (not total) porosity should be used.

(3) Distribution coefficients selected from the literature (instead of those obtained from tests on soils and geologic materials at the site) were used in the analysis.

Therefore, we did not agree with the applicants' analysis, and independently estimated the travel time and dilution factor. If a spill occurs, the closest public water supply would be Green Pond. We estimated travel times and dilution factors for both Green Pond and the nearest down-gradient well and determined that a spill flowing toward the nearest down-gradient well would be the critical case. The travel time to this well (No. 45 - PSAR, Figure 2.4.13-2), 1.6 miles southwest of the site, was conservatively estimated to be 20 years. The dilution factor was estimated to be about 29,200. The consequences of this postulated accidental spill are provided in Section 15.4 of this report.

2.4.5 Conclusions

Based upon our independent review and analyses, we conclude that an adequate water supply can be assured for safety-related purposes, and adequate flood protection can be provided for all safety-related facilities. Based on the analysis in Section 15.4, we also conclude that postulated accidental spills of radioactive liquids into the groundwater will not result in radionuclide concentrations in excess of the 10 CFR Part 20 limits at the nearest potable water supply.

2.5 Geology and Seismology

The seismology and geology review of this site addressed the geologic history of the region including physiographic, lithologic, stratigraphic and tectonic settings as well as the subregional and site-specific geology and seismology. In addition to reviewing data submitted in the Preliminary Safety Analysis Report, staff geologists and seismologists visited the site and its environs on two occasions. During those visits we examined the regional geology, bedrock exposures, and extensive core borings. We also conferred with local geological experts and the applicants' consultants concerning problems of geol gic interpretation in the site region. Some differences in detailed geologic interpretation of the site area may still exist between the applicants and local experts, but in our view these differences do not affect the site's suitability as the proposed location for the two-unit nuclear station.

A great deal of information has been gathered during the review of this site as well as other sites in the New England area. Since the regional aspects which also apply to this site have been addressed extensively in other reviews and safety evaluations, the main effort expended for this site dealt with resolving specific site issues which might have posed a hazard to safe operation of nuclear power plants at this location.

The particular items of concern at the Montague site were:

(1) The resolution of problems relating to the location and age of last movement of the "Triassic Border Fault".

(2) The potential for local surface faulting.

(3) The determination of the safe shutdown earthquake.

We are satisfied that investigations performed by the applicants have been sufficient to adequately assess site geologic conditions in accordance with "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A, 10 CFR Part 100. Based on our review of the available data and independent discussion with geologic experts familiar with the site region, the aforementioned concerns have been resolved. The references indicated in the geology and seismology sections are provided in Appendix B to this report.

2.5.1 Geology

2.5.1.1 Geologic Setting

Montague 1 and 2 are to be located near the junction of Wills Hill (a Triassic-Jurassic bedrock outcrop) and the Montague Plain (a delta formed in glacial Lake Hitchcock) approximately three miles east of Greenfield, Massachusetts. The 1900 acre site is about 60 miles north of Hartford, Connecticut, and about 75 miles west of Boston, Massachusetts. Geologically it is in the northern part of the Connecticut Valley Triassic-Jurassic Basin of the Piedmont-New England Tectonic Province. The site is situated on rocks of the Triassic-Jurassic Newark Group which in the site area consists of interbedded shales, sandstones, siltstones, conglomerates, and interbedded intrusive diabases and basalt flows. The development of the Triassic-Jurassic Basin and its relationship to adjacent structures provides the distinguishing geologic characteristic of the site area.

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2.5.1.2 Triassic-Jurassic Basin Faulting

The classical structural interpretation of the Connecticut Valley Triassic-Jurassic Basin evolved from the work by Emerson[1] and as summarized by Rodgers[2] indicates that a major normal fault existed at the eastern contact between the Triassic sediments and the Pre-Triassic crystalline rocks to the east. The applicants' early investigations indicated that evidence for a classical Triassic Border Fault did not exist and that the observed field relationships could be best explained by the existence of an unconformity. Additional investigations conducted during our review brought about a reasonable resolution of the alternative interpretations of the data as they relate to site safety. Three alternative hypotheses to the classical border fault concept have been presented by the applicants to describe the development of the basin: (1) downwarping, (2) development of one or more high angle faults beneath the basin ("basin-forming faults"), or (3) erosional control of the basins formation. As for the absolute validity of any of these hypotheses, it appears that at present there is insufficient data available to permit determination of a unique basin-forming mechanism. However, we consider that the most applicable hypothesis of the above as they relate to site safety is one requiring faulting for development of the basin. No large displacement fault has been found to date within the Triassic-Jurassic Basin itself. The apparent absence of major faulting and the relatively undisturbed stratigraphic relationship would indicate that last movement on the inferred "basin-forming" fault(s) would have occurred prior to late Triassic-Jurassic time (approximately 190 million years before present).

A major fault zone has been located approximately one mile east of the proposed site in the general vicinity of the mapped location of the classical Triassic border fault. Some disagreement remains as to the extent, sense of movement, age, and the total amount of displacement on this fault. However, based on K-Ar radiometric dating of fault zone material, no evidence of movements younger than Jurassic (190-136 million years before present) has been found along the mapped fault zone. The applicants' studies indicate that this fault is a major thrust on which movement is within predominantly Paleozoic crystalline terrain. It is contended that the development of this fault accompanied compressional activity during the Paleozoic followed by a larger episode of normal faulting. The applicants discuss the development of this fault extensively in the application and provide evidence such as that given below to support the above conclusions.

> "Small structures observed in cores of the mylonite and the quartz-muscovite shist indicate low angle reverse movement associated with the zone of cataclasis. The zone is interpreted to be a thrust fault. Later, small high angle, normal fractures are observed to cut across the reverse features and indicate a later tectonic event involving east-west extension. A detailed discussion of the small core structure is included in Appendix 2M (of the PSAR)."
>
> "Several radiometric age determinations were made for samples from the different lithologies in Area I. Samples of mylonite from the thrust zone give late Permian-early Triassic ages."

Rogers[2] in his section on the Connecticut Valley Synclinorium discusses the Ammonoosuc Fault as a postmetamorphic fault that can be traced by the offset of metamorphic isograds possibly thrusting lower grade rocks over older, higher grade rocks. This fault dips to the west at fairly low angles (averages 40 degrees). He also states,

> "It is also possible, however, that it is a normal fault, indeed, a large, late normal fault with similar attitude (although the dip is steeper) and similar silicification forms the eastern boundary of the Triassic basin in northern Massachusetts and can be traced into southern New Hampshire to within a few kilometers (miles) of the south end of the Ammonoosuc fault, and other late faults of the same kind are known in the intervening area and may interconnect them."

Other geologic experts familiar with the site region are not in complete agreement with the applicants' interpretation of the geology of the region especially as it relates to the existence of a "border fault." These experts maintain a strong view that this structure represents a fault with extensive normal displacement. It is clear to us, however, that the applicants have conducted an extensive investigation program to delineate and to date the time of last movement of all faulting in the site vicinity.

These investigations included detailed geologic mapping, an extensive core boring program, radiometric dating, structural and petrologic studies, and aeromagnetic and gravity studies. New observations and interpretations of the geology of the area, generated by the applicants' study, appear to require an alternative interpretation to that of the classical Triassic Border Fault hypothesis as being responsible for the formation of the northern Massachusetts section of the Connecticut Triassic-Jurassic Basin. It is difficult at this time, however, to completely rule out the possibility that this faulting is not representative of a Triassic Border Fault system of some type. The faulting located to date, however, has not indicated any evidence of movement since Triassic-Jurassic time (225-136 million years before present) and therefore cannot be considered capable in the meaning of "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A to 10 CFR Part 100.

2.5.1.3 Faulting

In addition to the existence of a "border fault," several other faults are mapped in the Triassic-Jurassic Basin within five miles of the site. These faults include the Falls River, Temple Woods, and other unnamed faults north of the general site area.

These faults are recognized because of offsets in the Deerfield diabase. A maximum offset of 800 feet for these faults occurs on the Falls River fault, the most prominent among them. The local faults are believed to be related to formation of the basin or fracture development in a flow basalt caused by compaction beneath it. The applicants describe the occurrence of these faults as they relate to the tectonic development of the site area as follows:

"Northeast trending dip-slip faults with small displacements were found to be common only in the Turners Falls (Falls River Fault) area. Goldstein[165] suggests that these faults formed contemporaneous with basin subsidence which tilted the strata and could have imposed normal motions on some of the northeast trending strike-slip faults. Drape folds are associated with the dip-slip faults; while other soft sediment folds are interpreted to have been caused by gravity sliding. These soft sediment structures indicate that basin subsidence occurred prior to complete lithification of the Turners Falls Sandstone. The age of the dip-slip faults appears consistent with Jurassic K-Ar dates obtained on gouge from two small high-angle faults at other locations (see Appendix 2N)."

"These small, high angle faults near Waterford, Connecticut yielded K-Ar ages, on illite, of 174 and 175 million years.[4]"

Since evidence indicates that these faults are probably related to the last major tectonic activity associated with the development of the Connecticut Triassic-Jurassic Basin (225-136 million years before present) and they are limited in extent, they are not considered capable nor hazardous to the safe operation of the Montague nuclear plants.

2.5.1.4 Lineaments

During the course of investigations for this site a number of lineaments were located in the site region by inspection of Earth Resources Technology Satellite imagery. Further evaluations of these linears indicated that they were most likely representative of rock jointing patterns and did not reflect the existence of faulting. Field checks in the area of a number of these linears indicated no offset in the bedrock.

2.5.1.5 Site Geology

The geologic conditions at the site are straightforward in that there is little structural complexity of the bedrock in the site area. The bedrock consists of interbedded Triassic-Jurassic sandstones and conglomerates, partially covered with Pleistocene tills and deltaic deposits. Extensive trenching to bedrock did not uncover evidence of faulting at the plant site.

The site bedrock exhibits a variable joint spacing from two inches to five feet. Three sets of joints are present: a bedding plane set striking N13°E, dipping 22°E, a vertical set striking N60°W, and a set striking N2°W, dipping 45°W. Numerous slickensides were recognized along favorably oriented joints and most likely formed during development of the synclinal structure within the northern end of the basin.

Based on our review of the data, we conclude that there is no geologic structure in the vicinity of the site that could cause surface displacement or tend to localize earthquakes at the site.

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2.5.1.6 Tectonic Provinces

The applicants contend that New England is not a single "Tectonic Province" as defined in Appendix A. They base this on their interpretation of the tectonic history and structural features of the region. Based primarily on the conclusions and methodology of Rodgers[2], the site region (200 mile radius) is subdivided into four tectonic provinces. These provinces are the New York, Western New England, Central New England, and South-eastern Provinces. Another province, the White Mountain Plutonic Province which is proposed to be more relevant in determination of the safe shutdown earthquake for the site is described by the applicants in the Preliminary Safety Analysis Report as follows:

> "The previously described provinces are related to Paleozoic orogenesis and are consistent with the conclusions and methodology of Rodgers.[5] A fifth province can be defined on the basis of Mesozoic tectonic effects which overprint part of the Paleozoic provinces. This province encompasses the region which has been intruded by alkalic plutonic and volcanic rocks of the White Mountain Magma Series. Radiometric age dating of these rocks has shown that the intrusive activity continued intermittently throughout the Mesozoic and that some members are as young as 96 m.y.[174,175] The major "belt" of intrusives extends north-west from the Maine-New Hampshire seacoast across central New Hampshire. The province boundaries have been further extended to the west to include the Ascutney and Cuttingsville Stocks in Vermont. These boundaries encompass all known plutonic bodies associated with the White Mountain Magma Series. The province trends offshore southeast of the New Hampshire coast and includes a magnetic high offshore from Cape Ann, Massachusetts. This anomaly is similar to anomalies associated with known plutons of the White Mountain Magma Series and is coincident with a pronounced topographic high on the continental shelf.[181]"

It should be noted that the applicants have recently reevaluated data on the Cuttingsville Stock and now consider it to be a Monteregian Hills type of pluton and not part of the White Mountain Magma Series or Plutonic Province.

In our review, we determined that the proposed site lies within the New England-Piedmont Tectonic Province based on larger scale provinces which are more in accord with those proposed by King,[3] Eardley,[4] Rodgers,[2] and Hadley and Devine[5] for eastern North America.

2.5.2 Seismology

2.5.2.1 Vibratory Ground Motion Summary

The historical earthquake activity in the site region appears generally lower than that for much of the remainder of New England. Capable faults have not been identified in the vicinity of the site. Based on these findings there is no reason to expect future earthquake activity to be localized in the site area. We conclude, based on the data reviewed, that the Montague site should be considered to be in the

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New England-Piedmont Tectonic Province. We have also concluded that a Modified Mercalli (MM) intensity of VII-VIII at the site represents a reasonable and appropriate safe shutdown earthquake in accord with Appendix A to 10 CFR Part 100. The trend of the mean of the peak vibratory ground acceleration values corresponding to this intensity is 0.2g. The spectral content of the vibratory ground motion is specified by the response spectra at free-field foundation level recommended in Regulatory Guide 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants. The vibratory ground acceleration corresponding to the operating basis earthquake is taken to be one-half that for the safe shutdown earthquake.

2.5.2.2 Tectonic Provinces

In arriving at the safe shutdown earthquake, the Montague site was considered to be located in the New England-Piedmont Tectonic Province. The New England part of the province, as it is regarded here, coincides with the northern Appalachians as described by King[3], Eardley[4], Rodgers[2], and Hadley and Devine.[5]

2.5.2.3 Maximum Earthquake

New England has one of the longest records of historical earthquake activity anywhere in the United States.[6] The Montague site is located in an area of New England which, based on historical accounts, is relatively quiet seismically. The nearest historical earthquakes reported in the vicinity of the site were identified on the basis of isolated felt reports at Amherst, Massachusetts, 8 miles south of the site, for which no intensity estimates have been given.[7] Several earthquakes, the largest having an intensity of VIII (MM)[6] (reclassified to be no greater than VII (MM), see below) have been reported within the Connecticut River Basin. However, these earthquakes have been concentrated mainly in the southern part of the Basin (more than 50 miles from the site), and there is no clear association between the earthquake activity and basin-forming faults of Triassic age near and along the margin of the basin.

We recognize that different regions of the New England-Piedmont Tectonic Province have exhibited different levels of historical earthquake activity. We have relied heavily on the historical record of seismicity in determining the seismic design for the Montague site. Several zones of relatively high seismic activity are discernible:[5,6,8,9]

(1) A zone extending from the Boston-Cape Ann, Massachusetts area northward and westward (Zone A).

(2) A broad region in the vicinity of Ottawa-Montreal (Zone B).

(3) The lower St. Lawrence River Valley northeast of Quebec (Zone C).

(4) A zone at the mouth of the Bay of Fundy (Zone D).

(5) A zone extending from the vicinity of New York City to the southern Connecticut River Valley northeast of New Haven (Zone E).

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The degree to which identification of these zones may have been influenced by population density is not clearly understood at this time. However, the larger earthquakes of New England, reported in the historical record,[6] have been located in these zones: (1) in Zone A, intensity VIII (MM) near Cape Ann in 1755, intensity VIII (MM) in Newbury in 1727, intensity VII-VIII (MM) at Woburn, Massachusetts, in 1817, and intensity VII (MM) at Lake Ossipee, New Hampshire, in 1940, (2) in Zone B, intensity IX (MM) near Montreal in 1732 and intensity VIII (MM) somewhat southeast of Ottawa in 1944, (3) in Zone C, several intensity IX (MM) earthquakes and an intensity X (MM) earthquake in 1663, (4) in Zone D, an intensity VIII (MM) earthquake in 1869, (5) in Zone E, intensity VII (MM) earthquakes at New York City in 1737 and 1884 and an intensity VIII (MM) just northeast of New Haven (East Haddam) in 1791. Experts from the Dominion Observatory in Canada responsible for assessing intensities of Canadian earthquakes, have recently investigated data from the 1869 earthquake in the Bay of Fundy and, on the basis of their studies, have reassessed the epicentral intensity to be no greater than VI (MM).[10] The intensity of the 1791 earthquake has been re-evaluated and assessed as a V-VI (MM) by Linehan.[11] Seismologists from the National Oceanic and Atmospheric Administration, U.S. Department of Commerce, have recommended that the 1791 earthquake be reclassified as intensity VII (MM).[12] In determining the Safe Shutdown Earthquake we have accepted these reassessments as accurate. Several isolated earthquakes have also been reported outside of these zones. Earthquakes in New England outside the zones of high activity have not had reported intensities greater than VII (MM).[5,6]

Geologic structure in Zone B is indicated by a set of young (Cretaceous) intrusive bodies which include the Monteregian Hills near Montreal. The earthquake activity in the vicinity of Ottawa-Montreal (Zone B) and in the lower St. Lawrence River Valley (Zone C) also coincides with a region of mapped, intense normal faulting, the Ottawa-Bonnechere graben and related faults and grabens extending along the St. Lawrence which have been identified as of post-Early Cretaceous age.[3,5,13] This feature marks the boundary between the northern Appalachian foldbelt and the Canadian shield. Much of the earthquake activity in Zone A is coincident with a zone of shallow intrusives and associated volcanic rocks of predominantly Jurassic-Cretaceous age. In determining the Safe Shutdown Earthquake for the Montague site, we recognize the association between the earthquake activity and structures in Zones A, B, and C. Based on this association, it is not assumed that earthquakes in the vicinity of Ottawa-Montreal (Zone B) or in the lower St. Lawrence Valley (Zone C) could occur near the site.

A more critical problem in the Montague review was the establishment of the minimum distance between the Montague site and the structures associated with the earthquake activity in Zone A and with the zone of intrusives in that area. We therefore required that the applicants provide a reasonable basis for determining the limits of this structural zone. In the Preliminary Safety Analysis Report the applicants have presented evidence for associating the earthquake activity with the zone of shallow intrusives, defined by the applicants as the White Mountain Plutonic Province, and have argued that future large earthquakes in the area can be reasonably expected to be confined to a region whose perimeter circumscribes the mapped and inferred shallow intrusive bodies. This zone extends to within about 60 miles of the Montague site. However, the historical earthquake activity extends somewhat beyond this zone to the southwest. To help resolve this problem, the applicants were requested to provide a map showing the seismic strain release for the area. The strain release associated with an intensity V (MM) earthquake was considered to represent background level for New England and values above this were contoured. The map shows that most of the seismic strain energy released in historical earthquakes coincides with the zone of intrusive bodies. Two lobes in the strain release contours extend to the south to distances as great as 30 miles from the zone of mapped intrusives but no closer than about 35 miles from the Montague site. Because the extent of the structures with which the earthquakes are associated is not clearly defined, we believe a more conservative boundary which includes both the intrusives and the historical seismic strain energy release is appropriate. An intensity VIII (MM) earthquake, corresponding to the largest earthquake in Zone A, assumed to occur at the nearest approach to the Montague site of the strain release contours - i.e., about 35 miles from the site - results in a site intensity of VII-VIII (MM).

2.5.2.4 Safe Shutdown Earthquake

The applicants have proposed to design Montague 1 and 2 for a safe shutdown earthquake acceleration of 0.2g.

In 1954 Neumann[14] developed an empirical relationship between earthquake intensity and ground acceleration. More recently Trifunac and Brady[15] have published a relation between intensity and acceleration which was developed using many additional observations. Trifunac and Brady's data essentially corroborate the relationship published by Neumann. Utilizing either the Neumann or the Trifunac-Brady relation between intensity and acceleration, the trend of the mean of the peak acceleration values corresponding to intensity VII-VIII (MM) is 0.2g.

Based on these considerations the applicants' proposed acceleration of 0.2g for the safe shutdown earthquake would adequately represent a site intensity of VII-VIII (MM). We believe this analysis is consistent with the guidelines of Appendix A of 10 CFR Part 100 and leads to the conclusion that 0.2g is a reasonable and appropriate acceleration at free-field foundation level representing the safe shutdown earthquake for use in the seismic design of Montague 1 and 2. The design response spectrum shape is specified by Regulatory Guide 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants.

2.5.2.5 Operating Basis Earthquake

The design vibratory ground acceleration for the operating basis earthquake is taken to be one-half the design vibratory ground acceleration for the safe shutdown earthquake consistent with the guidelines of Appendix A of 10 CFR Part 100.

2.5.3 Stability of Subsurface Materials

This section describes the criteria, evaluation of data, analyses and conclusions regarding the stability of foundation materials underneath the proposed nuclear power plant structures.

All soils below seismic Category I structures will be removed and excavations for all seismic Category I structures will be into bedrock. Concrete will be used as structural fill where required for foundation grading. All pockets of decomposed rock or other unsuitable material will be removed if encountered at foundation levels.

The bedrock in the site area consists of interbedded siltstones, sandstones and conglomerates. The Rock Quality Designation value is generally above 80 percent, and compressive strength exceeds 7,000 pounds per square inch. The average shear wave velocity is in excess of 5,000 feet per second. Granular backfill around seismic Category I structures will be compacted to at least a minimum relative density of 85 percent.

The quantity of ground water flow in the bedrock is small. Water pressure tests show that the intact rock has a very low permeability. Ground water seepage into the excavation will be small and can be handled by conventional ditch and sump methods. No heave problems are expected due to excavation and dewatering.

Because all seismic Category I structures will be founded on bedrock, settlement of these structures will be negligible. The maximum bearing pressure, 8,100 pounds per square foot, is only a small fraction of the allowable bearing pressure. Lateral earth pressures due to earthquakes will be computed using state-of-the-art procedures. Criteria for the minimum design factors of safety are 3.0 for bearing capacity and 1.2 for hydrostatic uplift. Based on these criteria, an evaluation of foundation data and the analyses contained in the application (including Supplement 5), the applicants have concluded that the bedrock is capable of supporting all loads that will be imposed by the power plant structures under both static and dynamic conditions.

We have reviewed the applicants' criteria, basis of evaluation, and analyses and have concluded that the applicants' approach is sufficiently conservative and that the bedrock will provide acceptable foundation support.

2.5.4 Slope Stability

This section describes the criteria, evaluation of data, analyses, and conclusions regarding the stability of all slopes, the failure of which could adversely affect the nuclear power plant.

The only natural slope in the immediate area of the site is Wills Hill northwest of the plant area. The maximum slope of the hill is 14°. The stability of the slope was analyzed using an assumed joint friction angle of 37° and continuity of all

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joints. There is no evidence to indicate that slides have occurred at Wills Hill in the past.

In addition to the above natural slope, a rock cut will be made between the fuel buildings and the standby service water cooling towers. Stability analyses were also made for this slope.

The proposed criteria for minimum design factors of safety are 1.5 for all static loading conditions and 1.2 for safe shutdown earthquake loading conditions for slope stability. Based on these criteria, evaluations of data and the analyses contained in the Preliminary Safety Analysis Report (including Supplement 4), we find that the proposed natural and cut site slopes will be stable under both static and dynamic loading conditions.

During construction, the applicants will inspect the excavated rock cuts and exposed surfaces will be mapped in detail. In addition, measurements of joint friction will be made in direct shear to confirm values used in the slope stability analyses. We will review the results of the inspection and measurements, along with final analyses, when the information is provided in the Final Safety Analysis Report.

The applicants conclude that, based on the evaluation of data and analyses contained in the application the slopes satisfy the proposed design criteria for slope stability. We conclude, based on the information presented, that the applicants' assumptions and criteria are sufficiently conservative.

The applicants have agreed to install four to six permanent benchmarks to allow for long-term monitoring of the stability of the rock cut above the fuel buildings and the standby service water cooling towers. We conclude that this monitoring program as proposed by the applicants is acceptable and will provide early indication of any instability of the rock cut.

In summary, based on the information available, the staff concludes that the applicants can design and construct the proposed power plant facilities to satisfy the foundation engineering requirements of 10 CFR Part 100.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Conformance with General Design Criteria

The applicants provided in Section 3.0 of the Preliminary Safety Analysis Report, an evaluation of the design bases for Montague 1 and 2 with respect to the NRC's General Design Criteria as contained in Appendix A to 10 CFR Part 50. Based on our review of the preliminary design and of the proposed design criteria, we conclude that upon resolution of the outstanding matters discussed in Section 1.8 of this report, Montague 1 and 2 can be designed, constructed and operated to meet the requirements of the General Design Criteria.

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

Except as identified below, structures, systems, and components important to safety that are required to be designed to withstand the effects of a Safe Shutdown Earthquake and remain functional have been properly classified as seismic Category I items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facility are designed to other than seismic Category I requirements including those portions of Category I systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of a Safe Shutdown Earthquake and remain functional have been identified in an acceptable manner in Tables 3.2.1-1 and 3.3.1-2 of the Preliminary Safety Analysis Report, except for the cooling lines identified below.

The applicants have classified the cooling lines to the reactor recirculation pumps as non-seismic Category I and Quality Group D; and in response to staff Request 211.1, the applicants provided the basis for their classification. The response states that analyses of reactor recirculation pump motor behavior following complete loss of cooling water indicates that if the initial cooling water loss alarm and the subsequent bearing temperature alarm (about 6 minutes later) are both ignored, the bearings will continue to operate another 6 to 10 minutes before they will suffer any damage. The response further states that such damage will not cause motor seizure and assuming the worst possible steel to steel friction, the motor will trip on overload caused by the added friction. Preliminary review of these results indicates that for this event the consequences of the cooling water failure would not result in fuel damage.

The staff plans further review with the applicants of the leakage characteristics of primary coolant through the pump seals as a result of assumed cooling water failure. As these studies progress, further information may be required and certain system changes could be indicated. We conclude that the potential nature of the system changes involved will permit this study and evaluation to be completed prior to a decision on issuance of construction permits. This matter is outstanding and will be resolved prior to issuance of a construction permit as indicated in Section 1.8.

The basis for acceptance in the staff's review has been conformance of the applicants' designs, design criteria, and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in General Design Criterion 2, and with Regulatory Guide 1.29 - Seismic Design Classification, technical staff positions, and industry standards.

Subject to the resolution of the matter cited above, we conclude that the proposed structures, systems and components that are important to safety and that are designed to withstand the effects of a safe shutdown earthquake and remain functional and that have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable regulatory guide, staff technical positions, and industry standards, are acceptable. Design of these items in accordance with seismic Category I requirements provides reasonable assurance that the proposed plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.2.2 System Quality Group Classification

Except as identified below, fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicants have applied a classification system (Safety Classes 1, 2, 3 and Other Structures, Systems and Components) which corresponds to the Commission's Quality Groups A, B, C and D in Regulatory Guide 1.26 - Quality Group Classifications and Standards for Water-Steam-and-Radioactive-Waste-Containing Components of Nuclear Power Plants to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary; (2) to permit shutdown of the reactor and maintenance of the safe shutdown conditions; and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2.1-1, 3.2.5-1 and 3.2.5-2 and on the system piping and instrumentation diagrams in the Preliminary Safety Analysis Report, except for the cooling lines discussed above in Section 3.2.1.

The basis for acceptance in the staff's review has been conformance of the applicants' designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's regulations as set forth in General Design Criterion 1, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, and to the provisions of Regulatory Guide 1.26, staff technical positions, and industry standards.

The staff concludes that fluid system pressure-retaining components important to safety that are designed, fabricated, erected and tested to quality standards in conformance with the Commission's regulations, the applicable Regulatory Guide, staff technical positions and industry standards are acceptable. Conformance with these requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

All the seismic Category I structures listed in Table 3.2.1-2 of the Preliminary Safety Analysis Report will be designed to withstand the effects of the design wind, and all seismic Category I systems and components located within these structures will thereby be protected from its effects. Category I systems and components located outside the structures and thus exposed to the wind, will be designed to withstand its effects.

The design wind specified for the plant has a velocity of 90 miles per hour at an elevation of 30 feet above grade and is based on a recurrence interval of 100 years. The basis for establishing these wind parameters was discussed in Section 2.3.1 of this report.

The procedures that will be used to transform the wind velocity into pressure loadings on structures, systems or components, and the associated distribution of wind pressures and drag coefficients will be in accordance with ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures." This document has been previously used and recognized and has been accepted for use by the NRC staff.

The procedures utilized to determine the loadings on seismic Category I structures induced by the design wind specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant's seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

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3.3.2 Tornado Loadings

All the plant structures whose failure can directly affect the safe shutdown of the plant during non-accident conditions, as listed in Section 3.3.2 of the Preliminary Safety Analysis Report, will be designed to withstand tornado effects; and all safety-related systems and components located within these structures will thereby be protected.

The design basis tornado specified for the plant has a tangential wind velocity of 290 miles per hour and a translational velocity of 70 miles per hour. The maximum pressure drop associated with the tornado is 3.0 pounds per square inch with a maximum pressure drop rate of 2.0 pounds per square inch per second. Further, an appropriate spectrum of tornado-generated missiles is also postulated as will be discussed in Section 3.5 of this report. The basis for selecting and establishing these tornado parameters was discussed in Section 2.3.1 of this report.

The procedures that will be used to transform the tornado wind velocity into pressure loadings on structures will be in accordance with the ANSI A58.1-1972 document, cited above, except that the pressure will be applied uniformly over the full height of the projected area of the structure and no gust factors will be applied. The structures will either be vented or designed for the pressure drop associated with the tornado. The tornado missiles effects will be determined using procedures to be discussed in Section 3.5 of this report. The total effect of the tornado on structures, systems and components will be determined by an appropriate combination of its individual effects.

Tornado-generated loads will be combined with other applicable loads as will be discussed in Section 3.8 of this report.

Structures that do not require hardening against the tornado will either be located such that their structural failure will not affect the integrity of structures that will be designed for the tornado, or will be designed not to collapse under the tornado wind load.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of a design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The facility yard grade and the ultimate heat sink cooling tower basins for the Montague 1 and 2 site will be located above the design basis flood level, including wave runup, and, therefore, will not be subject to flooding. The exterior walls and slabs of all safety-related structures extending below the ground level will be protected from groundwater by waterproofing. There will not be any exterior wall penetrations below finished grade into compartments which house safety-related equipment.

Seismic Category I structures will not be designed for flood forces because the site grade exceeds the elevation of the probable maximum flood from the Connecticut River. However, since the plant will be located at the base of Wills Hill, runoff water will be collected in an open channel at the base of the hill, and then drained into an underground drainage piping network. The required level of plant protection is discussed in Section 2.4.2.

As a result of our review, we conclude that the proposed water level (flood) design is in accordance with General Design Criterion 2 and Regulatory Guide 1.59 - Design Basis Floods for Nuclear Power Plants, and is acceptable.

3.4.2 Design Procedures

Certain natural phenomena, such as flood current, and wind wave, that are associated with environmental forces are not applicable to the Category I structures of this plant, since the finished grades around these structures are located above the probable maximum flood elevations.

The procedures described in the Preliminary Safety Analysis Report and utilized to determine the loadings on seismic Category I structures from the finished plant grade (approximately 340 feet mean sea level) to their foundations, assuming saturation to the top of grade, are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of floods, the structural integrity of the plant's seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures are adequately protected and may be expected to perform their intended safety function if needed. Conformance with these design procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.5 Missile Protection Criteria

3.5.1 Tornado Missiles

We have reviewed the information supplied in the application and in the applicants' letter dated August 1, 1975 concerning the analysis of tornado missile velocities and trajectories and find the results to be acceptable. The applicants have committed to require that the following spectrum of missiles (described in Safety Related Site Parameters for Nuclear Power Plants, WASH-1361) and impact velocities be used in the design of Montague 1 and 2.

Missile	Size*	Weight(pounds)	Velocity(feet per second)
A-Wood plank	4" x 12" x 12'	200	420
B-Steel pipe	3" Ø, 10' long, schedule 40	78	210
C-Steel rod	1" Ø x 3' long	81	310
D-Steel pipe	6" Ø, 15' long, schedule 40	285	210
E-Steel pipe	12" Ø, 15' long, schedule 40	743	210
F-Utility pole	13.5" Ø x 35' long	1490	210
G-Automobile	20 ft^2 frontal area	4000	100

*(")inches, (')feet, (Ø)diameter

The design must consider these missiles as striking in all directions. Missiles A, B, C, D, and E are to be considered at all elevations and Missiles F and G at all elevations up to 30 feet above all grade levels within 1/2-mile of the facility structures. We find this to be acceptable.

3.5.2 Missile Protection

The general design objective of missile protection is to ensure that structures, systems, and components, outside and inside containment whose failure could prevent safe shutdown of the plant, or result in significant uncontrolled release of radioactivity be protected against internally and externally generated missiles. Pressurized components and systems such as valve bonnets and hardware, retaining bolts, relief valve parts as well as high speed rotating machinery are considered potential sources of internally generated missiles.

Protection against these potential missiles will include orienting seismic Category I structures and components outside containment so as to minimize the probability of impact, providing missile barriers or shields, physically separating safety-related systems from non-safety related, and physically separating redundant components of safety-related systems so that a single missile could not damage both trains.

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As a result of our review, we conclude that the applicants' design criteria and bases for missile protection are in accordance with General Design Criterion 4 as it relates to structures housing essential systems, and Regulatory Guide 1.13 - Fuel Storage Facility Design Basis, as it relates to the design of the spent fuel pool system, and is in accordance with Regulatory Guide 1.27 - Ultimate Heat Sink for Nuclear Power Plants, as it relates to the design of heat sinks and connecting piping and are, therefore, acceptable.

3.5.3 Barrier Design Procedures

The analysis of structures, shields and barriers to determine the effects of missile impact, will be accomplished in two steps.

The barrier thickness required to prevent perforation by the missile will be determined in the first step. For steel barriers, the Stanford Research formula will be used. For concrete barriers the most conservative results obtained by the following formulas will be used:

(1) Petry formula.

(2) Ballistic Research Lab formula (BRLF).

(3) Combination of the Amman and Whitney formula and the National Defense Research Committee formula.

In the second step of the analysis, the overall structural response of the target when impacted by a missile will be determined using established methods of impactive analysis.

The load of the missile impact, whether the missile is environmentally generated or accidently generated within the plant, will be combined with other applicable loads.

All structures which contain safety related equipment will also be designed to withstand the loads and effects of a 15000 pound aircraft strike. The applicants plan to adopt the maximum allowable ductility ratios recommended in the Air Force Design Manual, "Principles and Practices for Design of Hardened Structures," AFSWC-62-138, December 1962. The applicants have indicated in Supplement 7 to the PSAR that the ductility ratio for any reinforced concrete flexural member shall not exceed 10. We find this criterion to be acceptable.

The procedures that will be utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the plant and the 15000 pound aircraft strike are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures or barriers are adequately resistant to and will withstand the effects of such forces.

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The use of these procedures provides reasonable assurance that the structural integrity of structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection in the event of design basis missile strikes. Seismic Category I systems and components protected by these structures will be, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.6 Criteria for Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

3.6.1 Outside Containment

General Design Criterion 4 requires that systems and components essential to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from postulated rupture of piping outside the containment. The systems and components essential to safety are those systems and components that are required to shut down the reactor and mitigate the consequences of a postulated piping failure without offsite power. The criteria require that for the postulated pipe failures, the resulting environmental effect will not preclude the habitability of the control room, or preclude the accessibility of other areas that must be occupied during an accident condition, or cause loss of function of electric power supplies, controls, and instrumentation needed to complete a safety action.

The applicants state that protection against pipe breaks outside containment will conform to the guidelines contained in Appendix A, General Information for Consideration of the Effects of a Piping System Branch Outside Containment, of the AEC letter from Mr. J. F. O'Leary, dated July 12, 1973.

To provide protection of essential systems and components, the plant design will accommodate the effects from postulated high energy system piping breaks with respect to blowdown jet and reactive forces, pipe whip, and environmental conditions resulting from the postulated pipe break. The applicants proposed a list, incorporating ten high energy piping systems, and identified the method of protection, i.e., separation, enclosure, or restraints, afforded. We agree with the contents of the list. The plant design will also accommodate the effects from postulated breaks in moderate energy systems, where necessary, and will use the possible combinations of physical separation, pipe whip restraints, and suitable enclosures to protect essential equipment in the event of a pipe break. The plant will be designed to withstand a high energy pipe break accident with whatever consequential damage that could occur, plus a single active failure, and still be capable of achieving a safe shutdown and maintaining the reactor in a cold shutdown condition.

As a result of our review, we conclude that the proposed protection against dynamic effects associated with the postulated failure of piping outside containment is acceptable.

3.6.2 Inside Containment

With respect to the systems located inside containment, the Preliminary Safety Analysis Report states that the criteria that will be employed for determining the systems which are to be evaluated, the locations and types of piping breaks which are postulated, and the protective measures against pipe whip to be provided, will be consistent with the requirements of Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

The methods of analysis described in the Preliminary Safety Analysis Report will adequately account for the dynamic loadings on systems, structures and components that are associated with pipe rupture assumptions. Use of these methods will provide adequate assurance that the containment structure, unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected.

For determination of restraint loading due to the postulated pipe ruptures, the applicants have committed to utilizing the dynamic analysis procedures described in GESSAR (Docket No. STN 50-447) for that piping included in the General Electric scope of supply.

These procedures will yield conservative results for the large clearance, large deformation restraints described in Section 3.6.B of the Preliminary Safety Analysis Report (i.e., a gap size of approximately six inches) when used with the thrust forces calculated in accordance with the relationship given in Section 3.6.B of the Preliminary Safety Analysis Report. Design limits proposed by the applicants in Section 3.6.3.1.5.1B of the Preliminary Safety Analysis Report for use in the design of the pipe whip restraints will result in deformation limits as conservative as our limits for all methods and all materials employed. The methods used for formulating the hydrodynamic forcing functions induced by pipe rupture and the dynamic analysis for the pipe whip motion provide an acceptable basis for restraint design.

For the determination of restraint loading resulting from postulated ruptures of piping in the Stone and Webster scope of supply (all piping not furnished by General Electric), the applicants have committed to use the dynamic analysis procedures consistent with those that are acceptable to the staff as delineated in Section 3.6.2 of the NRC Standard Review Plan, NUREG-75/087.

The criteria used for the identification, design, and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis in satisfying the applicable requirements of General Design Criteria 1, 2, 4, 14, 15, 31 and 32.

The provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide adequate assurance that, in the event of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of a design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function, assuming a single failure.

The applicants have stated in the Preliminary Safety Analysis Report that they will incorporate the criteria contained in the AEC letter from Mr. J. F. O'Leary, dated July 12, 1973, in their analysis for high energy line breaks inside containment. In implementing these criteria, the applicants will designate design basis break locations throughout all high energy piping systems. These postulated break locations will be chosen on the basis of highest relative stress, or significant changes in flexibility of the piping. The protection will be provided against the dynamic effects of postulated pipe breaks and discharging fluids in piping systems containing high energy fluids and located inside the containment will be adequate to prevent damage to structures, systems and components to the extent considered necessary to assure the maintenance of their structural integrity. Such protection provides reasonable assurance that the safe shutdown of the reactor can be accomplished and maintained, as needed.

In addition to the protection provided for high energy systems, for those piping systems that do not operate at sufficient temperature or pressure to be considered high energy systems, the applicants will postulate sufficient leakage cracks to assure that essential equipment and components are protected from spraying fluid, flooding and the consequent environmental conditions that may be developed.

The criteria proposed for the identification, design and analysis of high and moderate energy fluid lines inside containment where postulated breaks and cracks may occur constitute an acceptable design basis for satisfying the applicable requirements of General Design Criterion 4.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic design response spectra (Operating Basis Earthquake and Safe Shutdown Earthquake) applied in the design of seismic Category I structures, systems, and components comply with the provisions of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentages of critical damping values used in the seismic analysis of seismic Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants." The synthetic time history used for seismic design of Category I plant structures, systems and components will be adjusted in amplitude and frequency content to obtain the response spectra that envelop the design response spectra specified for the site.

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Conformance with the provisions of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs for seismic Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems and components to withstand seismic loadings.

3.7.2 Seismic Subsystem Analysis

Modal response spectrum and time history methods for multi-degree-of-freedom systems will form the bases for analyses of all major seismic Category I structures, systems and components. When the modal response spectrum method is used, governing response parameters will be combined by the square-root-of-the-sum-of-squares method to obtain the maximum response. The absolute sum of responses or the modified double sum method will be used for modes with closely-spaced frequencies.

Three components of seismic motion will be considered: two horizontal and one vertical. The total response to the three components of seismic motion will be obtained by this method.

Floor spectra inputs to be used for design and test verification of systems and components will be generated from the time history method. Effects on floor response spectra of expected variations of structural properties and damping will be accounted for by widening the response spectra peaks by at least ± 15 percent. Torsional effects and stability against overturning will be considered.

Soil-structure interaction will not be considered since all seismic Category I structures will be supported on sound rock or on concrete backfill. No seismic Category I dams will be used in this plant.

We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicants provide an acceptable basis for seismic design.

3.7.3 Seismic Instrumentation

The installation of the specified seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuation of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12-Instrumentation for Earthquakes. On this basis we conclude that the seismic instrumentation to be provided is acceptable.

3.8 Design of Category I Structures

3.8.1 Steel Containment

The reactor coolant system will be housed within a free-standing steel cylindrical shell topped with a torispherical dome and fixed at its bottom into a concrete mat covered with a liner plate. The steel containment will be enclosed by a reinforced concrete shield building. The containment will utilize the Mark III pressure suppression system which will be relied on to limit the containment pressure and temperature transients following a postulated loss-of-coolant accident (LOCA).

The steel containment including all penetrations will be designed, analyzed, fabricated, constructed, inspected and tested in accordance with the rules of Subsection NE of the ASME Boiler and Pressure Vessel Code Section III, Division 1.

The containment will be designed for all the various load combinations that are considered credible, including appropriate combinations of accident loads and seismic loads. In addition, the containment will be designed to withstand a post-LOCA flooded condition in conjunction with an Operating Basis Earthquake (OBE). Such a flooding condition may be required to recover the fuel from the reactor after a postulated LOCA.

The materials that will be used in the construction of the containment will satisfy the requirements of Article NE-2000 of Subsection NE of the ASME Section III Code. The bottom region of the drywell walls and support columns that will be submerged in the suppression pool will be lined with stainless steel.

After the completion of the construction and prior to operation, the containment will be subjected to a structural proof test.

The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loading and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and guides acceptable to the NRC staff.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the sp -fied conditions without impairment of structural integrity or safety function. A seismic Category I concrete shield building protects the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2, 4, 16, and 50.

3.8.2 Containment Interior Structures

The major containment interior structures include the drywell, the reactor pedestal and shield wall, the refueling pool and operating floor, and various other intermediate floors.

The drywell will be a reinforced concrete cylindrical structure with a flat roof approximately five feet thick, stiffened by two deep girders forming the refueling pool. It will completely enclose the reactor vessel and the recirculation system. Its primary function is to divert the steam released in a postulated LOCA to the suppression pool. Because of the importance of this function, upon which the proper functioning of the pressure suppression system depends, the NRC staff has required that the drywell be treated to a certain extent as would a containment structure. Accordingly, the design and analysis procedures and the loads and load combinations will be similar to what is normally used and accepted for concrete containments. The design and analysis procedures for the lower vent region of the drywell will be based on finite element techniques to account for the vents. In addition, the staff have required that the drywell be subjected to a structural proof test at the design pressure to verify the structural capability of the completed vessel. The applicants have committed to such a test. Guard pipes which form extensions to the drywell will be designed, constructed and tested in accordance with Subsection NE of the ASME Section III Code.

The other interior structures will also be designed for appropriate load combinations that are considered acceptable to the NRC staff.

The criteria that will be used in the design, analysis and construction of the containment's internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetimes are in conformance with established criteria, and with codes, standards and specifications that are acceptable to the NRC staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of their structural integrity or the performance of their required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.8.3 Other Seismic Category I Structures

Seismic Category I structures other than the containment and its internal structures will be built from structural steel and concrete. The structural components will consist of slabs, walls, beams and columns. The design method for concrete will follow that specified in the ACI-318 Code; and for steel it will follow the American Institute of Steel Construction specifications, with appropriate modifications requested by the staff to account for loading conditions peculiar to nuclear power plants.

The criteria that will be used in the analysis, design and construction of all the plant's seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime, are in conformance with established criteria, and with codes, standards, and specifications acceptable to the NRC staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents, the structures will withstand the specified design conditions without impairment of their structural integrity or the performance of their required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.8.4 Foundations

The steel containment, its interior structures and the shield building will be founded on a concrete mat. The mat will be analyzed to determine the effects of the various combinations of loads expected during the life of the plant. Foundations of other Category I structures will also be constructed of reinforced concrete mats. Such foundations will be designed in accordance with the ACI-318 Code with appropriate modifications to the loading criteria described in Section 3.8.5.5 of the Preliminary Safety Analysis Report.

The criteria that will be used in the analysis, design and construction of all the plant's seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime, conform with established criteria, and with codes, standards, and specifications acceptable to the NRC staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques;

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and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of their structural integrity or the performance of the required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Vibration Operational Test Program

The applicants have agreed to perform a preoperational piping vibrational and dynamic effects test program to confirm that dynamic loadings on piping from operational transient conditions have been properly accounted for in the design and analysis of piping systems and restraints classified as ASME Class 1 and 2 components. This program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips and operating modes associated with the design operational transients.

The tests, as planned, will develop loads similar to those experienced during reactor operation. A commitment to proceed with such a program constitutes an acceptable design basis at the construction permit stage of our review in fulfillment of the applicable requirements of NRC General Design Criterion 15.

3.9.1.2 Analysis and Tests of Mechanical Equipment

The applicants will perform dynamic testing and analysis to confirm that all seismic Category I mechanical equipment will function during and after an earthquake of magnitude up to and including the safe shutdown earthquake, and that all equipment support structures are adequately designed to withstand seismic disturbances.

Subjecting the equipment and their supports to these dynamic testing and analysis procedures provides reasonable assurance that in the event of an earthquake at the site, the Category I mechanical equipment as identified in the Preliminary Safety Analysis Report will continue to function during and after a seismic event, and the combined loading imposed on the equipment and their supports will not exceed applicable code allowable design stress and strain limits. Limiting the stresses of the supports under such loading combinations provides an acceptable basis for the design of the equipment supports to withstand the dynamic loads associated with seismic events, as well as operational vibratory loading conditions without gross loss of structural integrity.

Implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 14.

3.9.1.3 Preoperational Vibration Assurance Program for Reactor Internals

With regard to flow-induced vibration testing of reactor internals, the applicants have stated in the Preliminary Safety Analysis Report that the first BWR 6 plant of each size will be considered a prototype design and will be instrumented and subjected to both cold and hot two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. From information contained in GESSAR (Docket No. STN 50-447), we note that the Perry 1 plant has been designated as the prototype applicable to BWR/6 plants of the size of Montague 1 and 2.

The preoperational vibration assurance program as planned for the Montague reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions that will be comparable to those experienced during operation. The combination of tests, predictive analysis and post-test inspection provide adequate assurance that the reactor internals may be expected, during their service lifetime, to withstand the flow-induced vibrations of reactor operations without loss of structural integrity. The continued integrity of the reactor internals in service is essential to assure the retention of all reactor fuel assemblies in their place as well as to permit unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

The conduct of the preoperational vibration tests constitutes an acceptable basis for demonstrating design adequacy of the reactor internals in fulfilling the applicable requirements of General Design Criteria 1 and 4 and in conforming with the provisions of Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals."

3.9.1.4 Analysis Methods for LOCA Loadings

The structural design adequacy of the reactor internals, including the control rod assemblies, will be confirmed by the applicants using a dynamic analysis of the reactor internals, together with the loads generated from the unbroken piping loops. This analysis will be performed under the combined effects of the postulated occurrence of a design basis loss-of-coolant accident and a Safe Shutdown Earthquake (SSE) including an SSE and a steam line break.

At the final design phase of our review of GESSAR (Docket No. STN 50-447), the staff will require a more detailed description of the analysis and results for the combined effects of a Safe Shutdown Earthquake and a steam line break. When this requirement has been satisfied, the dynamic system analysis which will be performed, will provide an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a recirculation line or a steam line break coincident with a Safe Shutdown Earthquake. The analysis will provide adequate assurance that the combined stresses and strains in the components of the reactor coolant systems and reactor internals, for these faulted conditions, will not exceed the allowable design stress and strain limits of ASME Section III, Appendix F (faulted limits) for the

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materials of construction, and that the resulting deflections or displacements of any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The assurance of structural integrity of the reactor internals under a recirculation line break or a steam line rupture concurrent with the most adverse loading event (SSE) provides added confidence that the design may be expected to withstand a spectrum of lesser pipe breaks and seismic loading events. Compliance with the dynamic system analysis and acceptance criteria listed above, constitutes an acceptable basis for satisfying the requirements of NRC General Design Criteria 2 and 4. A topical report providing the analysis and loads on reactor internals will be submitted by General Electric about mid-1976. This matter will be resolved and addressed in a supplement to this safety analysis report prior to a decision on issuance of the Montague 1 and 2 construction permits.

3.9.2 ASME Code Class 2 and 3 Components

All safety-related ASME Code Class 2 and 3 systems, components and equipment will be designed to sustain normal loads, anticipated transients, the Operating Basis Earthquake and the Safe Shutdown Earthquake within design limits which are consistent with those outlined in NRC Regulatory Guide 1.48, "Design Limits and Loading Conditions." The specified design basis loading combinations as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I provide reasonable assurance that in the event an earthquake should occur at the site or other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components are expected to remain within the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The applicants' design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components constitute an acceptable basis for design in satisfying NRC General Design Criteria 1, 2 and 4 and are consistent with recent NRC staff positions.

The criteria used in developing the design and mounting of ASME Class 2 and 3 safety and relief valves provides adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

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The criteria used for the design and installation of ASME Class 2 and 3 overpressure relief devices constitute an acceptable design basis in meeting the applicable requirements of NRC General Design Criteria 1, 2, 4, and 15 and are consistent with those specified in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

3.9.3 Component Operability Assurance Program - Active Pumps and Valves (ASME Code Class 2 and 3)

The applicants have described in the Preliminary Safety Analysis Report their program to assure the operability of active components, which is acceptable. Active components are defined as those pumps required to function and valves required to open or close or close during or following the specified plant condition.

The conduct of the applicants' proposed operability assurance program will provide adequate assurance of capability of active pumps and valves in seismic Category I systems including those which may be classified as ASME Code Class 1, 2, and 3, to withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and to perform the "active" function (i.e., pump operation, valve closure or opening) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. The specified component operability assurance procedures constitute an acceptable basis for implementing the requirements of General Design Criteria 1, 2 and 3 for active pumps and valves.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Operability of the instrumentation and electrical equipment is essential to assure the capability of such equipment to initiate protective actions in the event of a Safe Shutdown Earthquake (SSE) as necessary for the operation of engineered safety features and standby power systems. This matter is discussed below.

3.10.1 Stone & Webster Scope of Supply

The seismic qualification testing program which will be implemented for seismic Category I instrumentation and electrical equipment provides adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. The applicants referenced IEEE Standard 344, 1971 for the seismic qualification of Category I electrical equipment, and in addition, their program contains features which recognize and provide solutions for standard test program implementation problems, consistent with Standard Review Plan, Section 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment." This program constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

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3.10.2 General Electric Scope of Supply

Proper functioning of Category I instrumentation and electrical equipment is essential to assure the capability of such equipment to initiate protective actions in the event of a Safe Shutdown Earthquake (SSE) including, for example, operation of engineered safety features and standby power systems. The information presented in GESSAR (Docket No. STN 50-447) is consistent with Standard Review Plan, Section 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment." Our review of the General Electric seismic qualification program has been completed, however, documentation of our review and acceptance is outstanding. This documentation will be completed prior to a decision for issuance of Montague 1 & 2 construction permits.

The seismic qualification testing program to be implemented for seismic Category I instrumentation and electrical equipment will provided adequate assurance that such equipment may be expected to function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. This program constitutes an acceptable basis for satisfying the applicable requirements of NRC General Design Criterion 2.

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4.0 REACTOR

4.1 General

The nuclear steam supply systems for Montague 1 and 2 include the General Electric Company (GE) boiling water reactors (BWR) which generate steam for direct use in the steam-driven turbine generators. The design of the Montague 1 and 2 reactor is similar to that of the GESSAR-238 reactor and to the reactors for the Grand Gulf Nuclear Station, Units 1 and 2 (Docket Nos. 50-416/417) and the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440/441). The reactor design for these other applications has been reviewed and found acceptable by the NRC staff.

The fuel and heat source consists of slightly enriched uranium dioxide pellets contained in sealed zirconium alloy tubes of about one-half inch in diameter. These fuel rods, which are over twelve feet long, are assembled into fuel assemblies, each consisting of 63 fuel rods and one water-filled rod in an 8 x 8 array within a square open-ended zirconium channel box. Seven hundred and thirty-two of these fuel assemblies form a roughly cylindrical core.

The core is supported in a domed cylindrical shroud inside the reactor vessel. Steam separators and dryers are mounted on the shroud dome. Two external, motor-driven, constant speed recirculating pumps inject high-velocity water into the 20 jet pumps which are located in the annulus between the shroud and the reactor vessel. The high velocity water from the jet nozzles entrains and imparts energy to additional water from the annular region. The combined flow enters the bottom of the reactor core and boils as it passes upward through the fuel assemblies.

The steam which emerges from the core is separated from the steam-water mixture by the steam separators and dryers. The steam flows to the turbine-generator through four 26-inch diameter main steam lines. The heated condensate returns to the reactor through two 24-inch diameter feedwater lines and is injected into the annulus between the shroud and the vessel.

Control of the fission reaction rate within the core is achieved by the movement of neutron absorbing, cruciform-shaped control rods, and by variation of the flow rate through the core, thereby changing the steam fraction and moderator density. Individual hydraulic drives permit the control rods to be axially inserted from below the core to any degree desired or to be inserted fully and swiftly upon receipt of a trip signal (scram). Core flow rate is varied by the flow control valves in the recirculation lines.

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4.2 Mechanical Design

4.2.1 Fuel Mechanical Design

The 8 x 8 fuel assembly consists of 63 fuel rods and one unfueled, spacer-capture rod in a square 8 x 8 array within a square channel box. The rods are spaced and supported at the top and bottom by stainless steel tie plates. The rods are also held in alignment by spacer grids located along the assembly. The cladding is made of fully annealed Zircaloy and each rod contains a hydrogen getter. The fuel pellet is a right circular cylinder whose height to diameter ratio is approximately unity. It is chamfered and undished and made of uranium dioxide at approximately 94 percent of theoretical density. Gadolinia bearing pellets are used in the highest enriched rods which are distinguished from the rest of the fuel rods by means of an extended end plug design. A Zircaloy channel box contains the fuel assembly and is a load carrying member. It also provides a channel for coolant flow and control rod movement. The channel wall thickness has been increased from the previous 7 x 7 design. The benefit of the smaller diameter rod design of the new 8 x 8 fuel, is to reduce the thermal performance requirements of the fuel and minimize fuel-pellet mechanical interaction by use of the chamfered pellets, reduced pellet lengths and annealed Zircaloy cladding. Some of the main mechanical dimensions and parameters are given in Table 4.2.1 of this report.

The safety considerations in fuel assembly design are maintenance of basic assembly geometry for adequate coolant passage and preservation of cladding integrity to contain the fission products within the fuel rod.

In Section 4.2.1.3.5 of the Preliminary Safety Analysis Report, the applicants describe the loadings and design limits of the fuel assembly and cladding. They discuss the engineering design limits in terms of stress, strain, deflection, fatigue life and creep rupture. In addition, analytical methods to be used to demonstrate design adequacy are described. Such material properties as cladding yield and ultimate stresses, and other thermal properties are given. We reviewed these design bases in detail and found that they provide an acceptable description of design bases for the 8 x 8 fuel assembly. Details of our evaluation of the 8 x 8 reload fuel design are included in Appendix D of the GESSAR Safety Evaluation Report, NUREG-75/110 dated December 1975. The only differences between the BWR-6 8 x 8 fuel and the reload 8 x 8 fuel are that the total active fuel length is four inches greater in the BWR-6 fuel and the fission gas plenum length is 0.75 inches greater for the BWR-6 rods. These changes are not significant enough to change our general conclusions regarding 8 x 8 reload fuel given in Appendix D of the GESSAR Safety Evaluation Report, NUREG-75/110. In this report, we indicated that the nuclear design of the 8 x 8 reload fuel assemblies was reviewed by comparing its properties with those of equivalent 7 x 7 fuel assemblies and concluded the nuclear design of the 8 x 8 reload assemblies is acceptable.

General Electric performed mechanical tests which included: fuel assembly handling and shipping tests, channel box removal and replacement tests, water lug shear tests

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TABLE 4.2.1

FUEL ASSEMBLY DIMENSIONS

Fuel Assembly Data	
Overall length, inches	176
Nominal active fuel length, inches	148
Fuel rod pitch, inches	0.640
Space between fuel rods, inches	0.147
Channel wall thickness, inches	0.120
Fuel bundle heat transfer area, square feet	100.3
Fuel Rod Data	
Outside diameter, inches	0.493
Cladding thickness, inches	0.034
Pellet outside diameter, inches	0.416
Fission gas plenum length, inches	12.00
Pellet immersion density, grams/cubic centimeter	10.42

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and fuel assembly bending stiffness tests. These tests verified the ability of the fuel to be handled with no damage.

General Electric has accumulated extensive fuel operating experience with fuel having a range of design parameters that envelopes those of the 8 x 8 fuel. Although the design of the unfueled spacer-capture rods is new, it is based on experience with similar designs. Fuel assemblies with eccentrically located spacer capture rods have been successfully operated in the Humboldt Bay reactor.

The methods used by General Electric to calculate the effects of fuel pellet densification have been previously submitted in Topical Report NEDM-10735. This report has been reviewed and found acceptable by the NRC staff.

To replace the poison curtain previously used in BWRs, UO_2-Gd_2O_3 rods are introduced into the high enrichment assemblies. The thermal conductivity of such rods is slightly lower than that of the UO_2 rods. However, these rods are expected to operate at relatively lower power than a UO_2 rod. A different end plug design is used to distinguish them from other fuel rods. We have previously reviewed the use of UO_2-Gd_2O_3 rods and found them to be acceptable.

General Electric has plans to perform a test of the 8 x 8 design spacer grid and spacer-water rod locking arrangement. In addition, General Electric has a fuel surveillance program which is to be conducted on preselected 8 x 8 fuel assemblies during refueling outages of operating plants.

We plan to review the results of the above surveillance program during our review of the Montague 1 and 2 Final Safety Analysis Report. Also, a stress report should be provided for each component together with safety margin.

A calculation of cladding strain based on an empirical formula together with gross diameter measurements taken from an irradiated rod burst test was submitted in a Topical Report (NEDO-10505, May 1972). An update of this topical report as well as a re-analysis and evaluation will be required as new information becomes available.

Based upon the above testing and operating experience, we conclude that the proposed mechanical design criteria for the fuel are acceptable for Montague 1 and 2 at this construction permit stage of our review.

4.2.2 Reactivity Control Systems

Reactor power level can be controlled either by movement of control rods or by variation of the reactor coolant recirculation system flow rate. Certain fuel rods will also contain full length and others partial length gadolinium oxide, a burnable poison, to supplement the moveable control rods in controlling the core reactivity throughout the core life. A standby liquid control system is also provided as a backup shutdown system.

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Control rods (177 in number) are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate devices for control and for rapid insertion (scram). The drives have a common supply pump (and one parallel spare pump) as the hydraulic pressure source for normal operation and a common discharge volume for scram operation.

A control-rod-ejection accident, to be distinguished from the rod drop accident, is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced, so that any resulting nuclear transient would not be sufficient to cause fuel rod failure.

As indicated above, reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system can automatically adjust reactor power level to station load demand whenever the reactor is operating between approximately 65 percent and 100 percent of rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor power. This method of reactor power level control has been satisfactorily demonstrated in other reactors.

The standby liquid control system is available to pump a sodium pentaborate solution into the reactor vessel. See Section 4.3.3 of this report for further discussion of this system. This system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon decay.

On the basis of our review of the control rod, flow control and standby liquid control systems design, and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that these systems will satisfy the functional performance requirements and are acceptable. The details of the proposed design for the new rod pattern control system, which will allow use of ganged rod motion, have not yet been submitted by General Electric for our review. See Section 7.6.1 for further discussion of this system.

4.3 Nuclear Design

The BWR-6 reactor core for each unit of Montague 1 and 2 consists of 732 fuel assemblies and 177 control rods, arranged as shown in Figure 4.1-1 of the Preliminary Safety Analysis Report. A planar view of the fuel lattice cell is shown in Figure 4.2-3 of the Preliminary Safety Analysis Report. The fuel lattice cell consists of four square fuel assemblies and a cruciform shaped control rod. A fuel assembly has an 8 x 8 square array of rods, 63 of which are fuel rods; the 64th rod is a water-spacer rod. The cruciform shaped control rods contain 76 stainless steel tubes (19 tubes in each wing of the cruciform) filled with vibration compacted boron carbide powder. Moderator/coolant water occupies all space not taken up by fuel rods, control rods, and structural material. All of the water gaps between fuel assemblies are of the same size. Some of the water gaps, which do not include a control rod, are provided with guide tubes for both fixed and movable neutron flux detectors. Guide tubes are located in the space near the corners of two adjacent fuel assemblies.

There are a number of noteworthy features of the fuel lattice cell which are applicable to the first fuel cycle. These are: (1) the fuel rods are of four different uranium-235 enrichments, (2) the average enrichment of the uranium-235 isotope in the fuel bundle is 2.07 percent by weight, and (3) a number of fuel rods will incorporate an axially varying distribution of gadolinia.

We have reviewed and evaluated the nuclear design bases for the Montague reactors. The design bases consist of both safety design bases and power generation design bases. The general requirements of the safety design bases are: (1) that sufficient negative reactivity feedback be provided to prevent fuel damage as a result of abnormal operational transients; (2) that nuclear characteristics as required be exhibited to assure that the reactor has no inherent tendency toward divergent or limit cycle operation; and (3) that the excess reactivity of the core be limited sufficiently to assure that the reactivity control systems are capable of making the reactor subcritical with the highest worth control rod fully withdrawn. The general requirements of the power generation design bases are: (1) that sufficient reactivity be provided to reach the desired burnup for full power operation; (2) that continuous, stable regulation of core excess reactivity be allowed; and (3) that sufficient negative reactivity feedback be provided to facilitate normal maneuvering and control.

In addition to the general safety and power generation bases, the Montague 1 and 2 units are designed to meet a number of specific design bases. These are listed below:

(1) The power reactivity coefficient must always be negative.

(2) The moderator void reactivity coefficient must be negative.

(3) The Doppler reactivity coefficient must be negative.

(4) Control rod operating patterns and withdrawal sequences must be specified so that individual control rod worths are sufficiently low to prevent damage to the reactor system in the event of a rod drop accident.

(5) The maximum control rod withdrawal speed must not be greater than 3.6 inches per second.

(6) Control rod withdrawal increments must be limited so that a rod movement of one increment does not result in a reactor period which cannot be managed by an operator.

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(7) The power generation rate must be controlled so that the linear heat generation rate of 13.4 kilowatts per foot is not exceeded and so that the minimum critical power ratio is not less than 1.21, the operating limit for the plant.

(8) Sufficient burnable poison must be included in the nuclear design to ensure that the shutdown requirements can be met throughout the core life.

(9) The backup shutdown system must be capable of making the reactor subcritical at a temperature of 20 degrees Centigrade. It must be capable of inserting at least 600 parts per million of natural boron between a minimum rate and a maximum rate of 6 to 25 parts per million per minute.

Based on our review, we conclude that the nuclear design bases are acceptable, since they are in conformance with General Design Criteria 10, 11, 12, 26 and 27.

4.3.1 Power Distribution

We have reviewed and evaluated the information presented on power distribution. The power distribution is a function not only of the nuclear design, but also of the reactor operating state. Consequently, an infinite number of power distributions are possible for the Montague 1 and 2 reactors. Constraints are placed on the power distribution in order to limit the linear heat generation rate to less than 13.4 kilowatts per foot and to keep the minimum critical power ratio above 1.21, the operating limit. Target peaking factors for these design limits are given in Table 4.3.1 of this report. The operating conditions are periodically monitored to ensure compliance with the design limits.

The incore neutron monitoring system is composed of the Source Range Monitoring subsystem, the Intermediate Range Monitoring subsystem, the Local Power Range Monitoring subsystem, the Average Power Range Monitoring subsystem, and the Traversing Incore Probe subsystem. The Startup Range Monitoring range varies from the minimum source power level to about 10^{-3} percent of full power. The Intermediate Range Monitoring cover from 10^{-4} to 20 percent of full power. The Local Power Range Monitoring range varies from a few percent to 150 percent of full power. The Average Power Range Monitoring provide a continuous indication of average reactor power from a few percent to 150 percent of rated reactor power. The Average Power Range Monitoring subsystem is based on a subset of the Local Power Range Monitoring detectors. The Traversing Incore Probe subsystem is used to calibrate the Local Power Range Monitoring and to provide detailed data on axial flux distributions.

A discussion of power distributions in boiling water reactors is given in Appendix 4A of the Preliminary Safety Analysis Report. Appendix 4A indicates that the General Electric design methods are capable of adequately representing operating reactor states. The design methods are compared with measured data for both gross and local power distributions. The effect on power distributions of rod patterns, fuel burnup, flow variations, void distribution, xenon, hot and cold reactor conditions, and load following are discussed. The errors and uncertainties associated with the analytical

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TABLE 4.3.1

NUCLEAR DATA SUMMARY

Design Peaking Factor	
Maximum Fuel Bundle to Average Fuel Bundle	1.40
Axial Peak-to-Average	1.40
Local Peak-to-Average	1.13
Total Peak-to-Average	2.22
Water-to-Fuel Volume Ratio	2.50
Uranium Weight per Bundle (pounds)	415
Maximum Core Reactivity, All Rods in (K_{eff})	<0.965
Maximum Core Reactivity, Strongest Rod Out (K_{eff})	<0.99
Reactivity of Movable Control Rods, Cold (ΔK)	0.17
Range of Reactivity Coefficients	
Fuel Doppler Coefficient ($\Delta k/k/°F$)	-1.2 to -1.3 x 10^{-5}
Moderator Void Coefficient ($\Delta k/k/\%$ void)	-1.0 to -1.6 x 10^{-3}

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methods are also discussed and have been accounted for in the evaluation of fuel performance with the process computer.

We conclude that the discussions of the power distribution in Section 4.3 and in Appendix 4A of the PSAR are acceptable provided that questions and concerns arising from the staff review of Appendix 4A on the GESSAR docket are satisfactorily resolved. These questions a-e directed toward the statistical analysis of reactor data in establishing and accounting for errors and uncertainties. These questions are being addressed in Topical Report NEDO-20340 and resolution of our concerns will be accomplished as a part of our review of that topical report. Resolution of this matter need not be completed prior to a decision on issuance of the Montague 1 and 2 construction permit but will be supplied in the Final Safety Analysis Report.

We conclude that the information presented concerning the monitoring of powe. distributions is a.ceptable and the matters discussed above can be provided in the Final Safety 'nalysis Report.

4.3.2 Reactivity Coefficients

We have reviewed and evaluated the information presented on reactivity coefficients. The most important reactivity coefficients which determine the stability and dynamic behavior of the Montague 1 and 2 reactors are the Doppler reactivity coefficient, the moderator void reactivity coefficient, and moderator temperature reactivity coefficient. The power rea.tivity coefficient, which is associated with stability to power oscillations due to xenon and other causes, is a function primarily of the Doppler and moderator void reactivity coefficients.

The Doppler reactivity coefficient is a reactivity change associated with the Doppler broadening of absorption resonances of a material and is caused by changes in temperature. The Doppler reactivity coefficient is negative for the Montague reactors. The absolute magnitude of the coefficient increases with both increasing moderator temperature and increasing void fraction because the resonance escape probability is inversely proportional to the water to fuel ratio. The Doppler reactivity coefficient also becomes more negative as a function of fuel burnup due to the buildup of plutonium isotopes. Values of the Doppler reactivity coefficient are given in Table 4.3.1 of this report. In various transient analyses, the Doppler reactivity coefficient is taken to be -0.126 cents per degree Fahrenheit and is multiplied by a design conservatism factor of 0.9.

The Montague 1 and 2 reactors have a large negative moderator void coefficient of reactivity and a moderator temperature coefficient of reactivity which is much smaller in magnitude. The coefficients are obtained from partial derivatives of the infinite multiplication factor and neutron leakage as a function of control fraction* with

*The control fraction is defined as the ratio of the length of control rods inserted into the reactor to the total inserted length of all of the control rods.

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respect to the variables of temperature or void content with the reactor near critical. Of the two, the moderator temperature coefficient is less significant and plays a role only near the inlet region of a hot operating reactor where the void content is smallest. This coefficient may become slightly positive near the end of the fuel cycle. The strong moderator void coefficient of reactivity, on the other hand, gives the Montague reactors a number of important characteristics such as: (1) the capability of using coolant flow control for load following; (2) the inherent ability to self-flatten the radial power distribution; and (3) stability to xenon induced spatial power oscillations. Values of the void reactivity coefficient are given in Table 4.3.1 of this report. In various transient analyses, the moderator void coefficient is taken to be -11.5 cents per percent void fraction and is multiplied by a design conservatism factor ranging in value from 0.9 to 1.25.

We have reviewed this information and conclude that the discussion in the Preliminary Safety Analysis Report of the reactivity coefficients is acceptable. We find that the important prompt (Doppler) and void reactivity coefficients are negative throughout a fuel cycle. We further conclude that the absolute magnitudes of these coefficients are sufficiently large to ensure the stability of the Montague 1 and 2 reactors during power operation.

4.3.3 Control Requirements and Control

We have reviewed and evaluated the information presented on control requirements and control. The excess reactivity designed into the initial core is controlled by a control rod system supplemented by the use of a burnable poison, gadolinia, in a number of fuel rods. The gadolinia is uniformly distributed in a UO_2 fuel pellet but has an axial distribution within a fuel rod. The reactor is designed to permit the energy extraction of 12,000 to 19,000 megawatt days/ton averaged over the initial core loading and depending on the initial uranium enrichment. The excess reactivity is needed to compensate for reactivity losses due to moderator heating and boiling, fuel temperature increases, equilibrium and peak xenon, samarium poisoning, fuel depletion, and other low cross section fission product poisons.

The control rods provide a number of important operating functions. They are a means for: (1) rapidly decreasing the core reactivity during a reactor trip by being driven into the core; (2) bringing the reactor into the power operating range from either cold or hot shutdown conditions by planned rod withdrawal; (3) compensating for fuel depletion by planned rod withdrawal, and (4) shaping the power distribution by selective movement. The control rods are capable of shutting down the reactor ($K_{eff} < 1.0$) throughout the entire first fuel cycle for the most limiting condition, that is, for the reactor at 20 degrees Centigrade and for the highest worth control rod stuck out. The uncertainty associated with the calculation of the shutdown margin was estimated by General Electric to be about $0.005 \Delta K$.

Control rod withdrawal sequences are selected prior to operation in order to optimize core performance and to achieve low individual rod worths. The maximum controlled

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rate of reactivity addition during startup is 0.0011ΔK/sec. This value is based on the withdrawal of an in-sequence rod assuming a total rod worth of 0.01 ΔK, a peak incremental rod worth of 0.00033 ΔK/inch, and a maximum rod speed of 3.6 inches per second. Reactivity addition rates are considerably reduced at hot operating conditions from those under startup conditions due to the effects of void formation and redistribution as a rod is withdrawn.

The control of the reactor is not only dependent upon the movement of control rods but also upon changes which occur in various system parameters. Because the pressure changes caused by turbine load changes would bring about reactor power changes in a direction opposite to changes in turbine load, the reactor is operated as constant pressure device. The plant output is increased or decreased by changing the reactor circulating water flow and/or moving the control rods. As indicated previously, reactor startup from cold or hot conditions is accomplished by withdrawing control rods and keeping the recirculating water flow at a fixed value. The reactivity difference between the hot standby condition (5 percent power, 30 percent flow), as defined by General Electric, and the cold critical condition are 0.069 ΔK and 0.041 ΔK for beginning and end of cycle, respectively. These reactivity differences include temperature, void fraction, and xenon changes. By adjusting the recirculating water flow, the reactor power can be varied over approximately 35 percent of the power range. The power change produced by varying the recirculating water flow is nearly uniform and is based on curves developed during the reactor startup phase which correlate reactor power and flow for various control rod patterns. Control rod changes may also be made in the power range in conjunction with changes in the recirculating water flow; however, load following is usually accomplished by varying recirculating water flow. Spatial power disturbances, such as those caused by xenon redistribution, present no special control problem. The large negative power coefficient provides strong inherent damping of such disturbances or oscillations.

The Montague 1 and 2 reactors incorporate a standby liquid control system to satisfy the requirements of General Design Criterion 26. This system is capable of injecting a natural boron solution at the rate of 6 to 25 parts per million per minute and can bring the system coolant to a concentration of at least 600 parts per million. Based on the reactivity worth of the boron, this liquid control system, independently of any control rod action, is capable of shutting down the reactor to 20 degrees Centigrade from full power throughout the fuel cycle.

We conclude that the discussion of the control requirements and control is acceptable. We find that there is sufficient shutdown margin throughout the fuel cycle. We agree with the applicants that spatial power disturbances will be strongly damped by the large negative power coefficient. We conclude that power changes by control rod movement and/or changes in recirculating water flow can be made in an acceptable manner with respect to effects on the power distribution. We further conclude that adequate control of the excess reactivity exists throughout the fuel cycle. Finally, we conclude that a second shutdown control system requirement is satisfied by the standby liquid control system.

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4.3.4 Control Rod Patterns and Reactivity Worths

We have reviewed and evaluated the information presented on control rod patterns and reactivity worths. We find that specified rod withdrawal sequences are designed to limit rod worth so that the drop out of any control rod from the fully inserted position results in a peak fuel enthalpy of not more than 280 calories per gram. The selected rod pattern at any time will satisfy this requirement on the peak fuel enthalpy if the incremental control rod worth is restricted to no more than 0.01 ΔK even if the rod drop velocity reaches its maximum value of 2.79 feet per second.

As contrasted to other power producing reactors, the rod withdrawal sequences for a boiling water reactor are complex. In the startup range, the control rods are withdrawn to 50 percent control rod density leaving a checker-board pattern. Once a control rod has been selected for withdrawal in the startup range, it is withdrawn from its fully inserted to fully withdrawn position. The maximum in-sequence rod worth always occurs when the first control rod of an in-sequence group is withdrawn. The maximum out-of-sequence control rod worth will occur as follows: (1) all the control rods of an in-sequence group have been withdrawn; (2) a single rod from the next in-sequence group is withdrawn; and (3) the operator makes a single error by withdrawing the out-of-sequence control adjacent to the in-sequence control rod withdrawn in Step (2). The withdrawal of control rods during startup is performed in conjunction with permissives from the rod pattern control system, a system designed to preclude the withdrawal of out-of-sequence control rods. The maximum control rod worth, as controlled by the prescribed patterns in this system, are given in Figures 4.3-2a and 4.3-2b of the Preliminary Safety Analysis Report for two different levels of burnup.

In the power range, once a checkerboard control rod configuration has been achieved, the concept of in-sequence and out-of-sequence control rods is no longer meaningful since all interior control rods will have approximately the same reactivity worth. The worth of an interior control rod is about 1.5% $\Delta k/k$ in the hot operating state; however, the amount of reactivity which can be added due to a dropping control rod is restricted since only partial withdrawal of all the remaining rods in groups occurs. Control rod withdrawals in the power range are also restricted to limit the total power peaking factor. Control rod patterns are varied from time to time to maintain uniform burnup in each fuel assembly. In the power range the worst single operator error is defined as the selection and full withdrawal of the maximum worth control rod. This results in two methods for inserting potentially high reactivity into the reactor. The first method is by withdrawal of the high worth rod itself and the second method is by having an adjacent rod drive being completely withdrawn but with its control blades decoupled and remaining in the fully inserted position. This decoupled blade then falls out of the core.

In the startup range, the maximum in-sequence and out-of-sequence control rod worths are computed by means of full core, three group, two-dimensional, XY diffusion calculations. Homogenized cross sections are used for each fuel bundle. These cross sections are generated by using the General Electric standard lattice design methods for

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the controlled or uncontrolled fuel bundle. The effects of the axially distributed gadolinia are included in the XY diffusion calculations by using average cross sections and axial bucklings obtained from one-dimensional, three group, axial diffusion calculations.

In the power range, the control rod calculations are affected by the formation of steam voids in the moderator. The maximum control rod worth is calculated by means of three-dimensional XYZ diffusion theory for a control rod fully inserted or fully withdrawn for a constant void distribution. The initial void distribution is obtained from a three-dimensional coupled nuclear-thermal hydraulic calculation with the maximum worth out-of-sequence control rod fully inserted.

We conclude that the information presented on control rod patterns and reactivity worths is acceptable. Although the control rod patterns and withdrawal schemes are quite complex, we find that the rod pattern control system and the nuclear instrumentation can limit the worth of a control rod and the power peaking factor. Finally, we conclude that the restrictions on the rod patterns will limit the incremental control rod worth to approximately 0.01 ΔK and that no dropped rod would produce a peak fuel enthalpy of 280 calories per gram even if the rod were dropped at 2.79 feet per second.

4.3.5 Criticality of Fuel Assemblies

We have reviewed and evaluated the applicants' criticality analyses of the fuel assemblies. The criticality analyses were performed assuming a higher-than-normal average fuel enrichment and also assuming that there are no control rods or gadolinia. For the dry condition, the multiplication factor, K_{eff} is $<$ 0.50. In the fuel handling facilities, two fuel bundles give $K_{eff} \sim 0.74$, and four bundles $K_{eff} \sim 0.90$. Sixteen to twenty fresh fuel bundles with gadolinia present represent a critical array. Procedural controls prevent personnel from arranging four fuel bundles in a square array. See Section 9.1 of this report for further discussion of fuel criticality.

We conclude that the discussion on criticality of fuel assemblies is acceptable. We find that the procedural controls outlined in Section 4.3.2.7 of the Preliminary Safety Analysis Report are sufficient to prevent K_{eff} from exceeding 0.90 under normal conditions of fuel handling and storage and 0.95 for abnormal conditions.

4.3.6 Vessel Irradiation

We have reviewed and evaluated the information presented on reactor vessel irradiation. A one-dimensional, discrete ordinates transport code was used to calculate the neutron fluence at the pressure vessel assuming continuous reactor operation at rated power for 40 years. A radial power distribution representative of conditions throughout the life of the plant was used. Axial power distributions were calculated. The calculated fluence at the pressure vessel for neutrons of energies above one million electron volts is about 2.3×10^{18} neutrons/square centimeter.

From our review of the applicants' methods employed and described in Section 4.3.2.9 of the Preliminary Safety Analysis Report, we conclude that the calculated neutron fluence at the pressure vessel wall has been conservatively estimated.

4.3.7 Analytical Methods

The basic calculational procedures used by General Electric for generating neutron cross sections are part of its so-called Lattice Physics Model. In this model, the many-group fast and resonance energy cross sections are computed by a GAM-type of program. The fast energies are treated by multigroup integral collision probabilities to account for geometrical effects in fast fission. Resonance energy cross sections are calculated by using the intermediate resonance approximation with energy-and-position-dependent Dancoff factors included. The thermal cross sections are computed by a THERMOS-type of program. This program accounts for the spatially varying thermal spectrum throughout a fuel bundle. These calculations were performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperatures, burnup, voids, void history, the presence or absence of adjacent control rods, and gadolinia concentration and distribution in the fuel rods. As part of the Lattice Physics Model, three-group, two-dimensional XY diffusion calculations for one or four fuel bundles were performed. Local fuel rod powers, as well as single-bundle or four-bundle (with or without a control rod present) average cross sections, were calculated by this method.

The single or four bundle averaged neutron cross sections which were obtained from the Lattice Physics Model were used in either two- or three-dimensional diffusion calculations. Two-dimensional, XY calculations are usually performed in three-groups at a given axial location to obtain gross power distributions, reactivities, and average three group neutron cross sections for use in one-dimensional axial calculations. The three-dimensional diffusion calculations use 1.5 energy groups and can couple neutron and thermal hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes; however, at the design stage geometrical symmetry is used to reduce the size of the calculation. This three-dimensional calculation provides the best simulation of a boiling water reactor and yields gross three-dimensional power distributions, void distributions, control rod positions, reactivities, eigenvalues, and also average cross sections for use in the one-dimensional axial calculations.

The one-dimensional axial calculations are space-time diffusion calculations which are coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by General Electric with results obtained using the industry standard code, WIGLE.

The Doppler, moderator void, and moderator temperature reactivity coefficients were generated in a rudimentary manner from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt mode neutron lifetime were computed using the one-dimensional space-time code. The power coefficient was obtained by appropriately combining the moderator void, Doppler, and moderator temperature reactivity coefficients.

The response of a boiling water reactor (BWR) to any induced power oscillations is discussed in General Electric Topical Report APED-5652. The effect of spatially varying xenon concentrations on the stability of a BWR is specifically discussed in General Electric Topical Report APED-5640. These studies show that a BWR is stable to any xenon-induced power oscillations because of the damping effect of the large, negative, spatially varying void coefficient.

Appendix 4A of the Preliminary Safety Analysis Report gives a considerable amount of information on the comparison of calculated local and gross power distributions with measured data. The factors which influence the power distribution are discussed as well as uncertainties in the measurements and calculations. However, Section 4.3 of the Preliminary Safety Analysis Report does not provide any comparisons of calculations of K_{eff} with measured data for hot and cold conditions and with and without equilibrium xenon and samarium present. Comparison with experimental data of calculated control rod worths in the cold condition, shutdown margins for various conditions, the reactivity worths of the distributed gadolinia, and reactivity coefficients for various conditions is similarly lacking.

We conclude that the discussion of the analytical methods indicate that they represent the current state-of-the-art. We find acceptable the General Electric commitment on the GESSAR-238 Nuclear Island Standard Design, Docket No. STN 50-447, which also appears on the Montague docket as commitments by the applicants, to provide topical reports in the following areas:

(1) The lattice physics methods verification.

(2) Lattice physics methods verification.

(3) BWR simulator.

(4) BWR simulator methods verification.

(5) The void and Doppler reactivity coefficients.

We will require that the topical reports on the above five matters be reviewed and accepted prior to the decision on issuance of the Montague 1 and 2 construction permits.

4.3.8 Summary of Evaluation of Nuclear Design

The applicants have described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design. The staff concludes that the General Electric commitment on the GESSAR238 Nuclear Island Standard Design, Docket No. STN 50-447, which is also a commitment by the applicants on the Montague docket, to provide topical reports to demonstrate the ability of these analyses to predict reactivity and physics characteristics of the Montague plant is acceptable. Our review and acceptance of these topical reports will be completed prior to a decision on issuance of the Montague 1 and 2 construction permits.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicants have provided substantial information relating to core reactivity for the first cycle and have shown that means have been incorporated into the design to control excess reactivity at all times. The applicants have shown that sufficient control rod worth is available to shut down the reactor with at least a one percent $\Delta k/k$ subcritical margin in the cold condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

On the basis of our review, we conclude that the applicants' assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available during the operating license stage of our review. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28.

4.3.9 Control Rod System Structural Materials

The mechanical properties of structural materials selected for the control rod system components satisfy Appendix I of Section III of the ASME Code, and Part A of Section II of the Code, and the staff position that the yield strength assumed for cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch.

The controls imposed upon the austenitic stainless steel of the system conform to the provisions of Regulatory Guides 1.31, "Control of Stainless Steel Welding" and 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these provisions, as supplemented by information from the demonstration tests in progress, provide added assurance that stress corrosion cracking will not occur during the design life of the component. The compatibility of all materials used in the control rod system in contact with the coolant satisfies the criteria of NB-2160 and NB-3120 of Section III of the ASME Code. Both martensitic and precipitation-hardened stainless steels will be given tempering or aging treatment in accordance with staff positions. The provisions for cleaning and cleanliness control are in accordance with the provisions of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants."

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Conformance with the Codes, standards and Regulatory Guides mentioned above, and with the NRC staff positions on allowable maximum yield strength of cold worked austenitic stainless steel and minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for satisfying the requirements of General Design Criterion 26.

4.4 Thermal and Hydraulic Design

The thermal and hydraulic characteristics of the Montague 1 and 2 reactors are the same as those for the GESSAR-238 Nuclear Island Standard Design, Docket No. STN 50-447 design which has been reviewed at the Preliminary Design Approval stage by the NRC staff. The design basis for the core for steady-state operation, operational transients, or load-following maneuvers or abnormal transients are:

(1) No fuel damage.

(2) No undamped oscillations or other hydraulic instabilities.

(3) The maximum linear heat generation rate should not permit fuel centerline melting.

A summary of the thermal-hydraulic parameters for Montague 1 and 2 are given in Table 4.4.1 of this report.

The core and fuel design basis for steady-state operation are the minimum critical power ratio and linear heat generation rate. These limits have been defined to provide margin between the steady-state operating condition and any fuel damage condition to accommodate uncertainties and to assure that no fuel damage will result even during the worst anticipated transient condition at any time in life. Specifically, the minimum critical power ratio operating limit is specified such that at least 99.9 percent of the fuel rods in the core are expected not to experience boiling transition during the most severe abnormal operational transient. The steady-state operating limit for minimum critical power ratio is 1.21 and the peak linear heat generation rate is 13.4 kilowatts per foot.

GETAB, the General Electric Thermal Analysis Basis, is used to establish the thermal-hydraulic limits for the Montague reactors. The thermal-hydraulic parameter used for reactor design and operation is the critical power ratio, which is defined as the ratio of the critical bundle power to the operating bundle power. In GETAB, the uncertainties associated with the parameters affecting steady-state bundle power are treated statistically in order to satisfy the criterion that, during a transient, 99.9 percent of the rods in the core will not experience boiling transition.

Incipient center melting of the uranium dioxide pellet will occur in a linear heat generation rate range of 19 to 21 kilowatts per foot; this is higher than the peak linear heat generation rate during any abnormal operating transient.

TABLE 4.4.1

SUMMARY OF THERMAL HYDRAULIC PARAMETERS

Rated Power (megawatts thermal)	3579
Design Power (megawatts thermal)	3758
Steam Flow (10^6 pounds per hour)	15.396
Core Flow Rate (10^6 pounds per hour)	105
System Pressure (pounds per square inch atmosphere)	1040
Average Power Density (kilowatts per liter)	56
Maximum Linear Power (kilowatts per foot)	13.4
Maximum Heat Flux (British thermal units per hour square foot)	354,000
Maximum UO_2 Temperature (degree Fahrenheit)	3337
Core Inlet Enthalphy (British thermal unit per pound)	527.8
Total Peaking Factor	2.22

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Three types of stability are considered in the design of boiling water reactors. They are: (1) reactor core (reactivity) stability, (2) channel hydrodynamic stability and (3) total system stability. Two stability criteria are offered to demonstrate the stability of the system; the decay ratio, x_2/x_0, and the damping coefficient η. The decay ratio should be less than one and the damping coefficient greater than zero. The Montague Preliminary Safety Analysis Report presents values typical of a boiling water reactor.

The scope of our thermal-hydraulic design review included the design criteria and thermal-hydraulic performance. The applicants' thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

The staff concludes that the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable Regulatory Guides and staff technical positions and is considered acceptable for the construction permit stage of our review. In Section 4.4.3.5 of the Preliminary Safety Analysis Report, the applicants present typical values of stability and hydrodynamic performance and reference calculations that predate General Electric's introduction of the BWR-6 design.

4.5 Reactor Internals Materials

The materials for construction of components of the reactor internals have been identified by specification and found to be in conformance with the requirements of the ASME Code, Section III, Appendix I.

The materials for reactor internals that will be exposed to the reactor coolant have been identified and all of the materials are compatible with the expected BWR environment, as demonstrated by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible.

The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

Based on our review, we find that the controls imposed upon the fabrication of reactor internals constructed of stainless steel satisfy the provisions of Regulatory Guides 1.31, "Control of Stainless Steel Welding" and 1.44, "Control of the Use of Stainless Steel." Although the applicants take certain exceptions to these guides, the controls imposed on stainless steel welding, including the ferrite content of filler materials, have proven adequate for producing welds without evidence of fissuring. The applicants have agreed to demonstrate the adequacy of current welding controls by conducting tests to determine the ferrite content of production welds and to evaluate the degree of sensitization in welded type 304 and 316 stainless steel. Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these provisions provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service.

The use of the materials proven to be satisfactory by actual service experience and conformance with the provisions of these Regulatory Guides constitute an acceptable basis for satisfying the requirements of General Design Criteria 1 and 14.

4.6 Loose Parts Monitor System

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For the past few years we have required many applicants to initiate a program, or to participate in an ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable time. The applicants have indicated that they are following the General Electric program for development of a loose parts monitoring system and have committed to install a loose parts monitoring system in the Montague 1 and 2. We conclude this is acceptable for the construction permit stage of our review.

4.7 Gross Failed-Fuel Monitor

For some time we have been requiring that nuclear power plants include a system to permit detection of any potential gross fuel failures in the core. The purpose for such a system is that it would allow for corrective action following a postulated gross fuel failure to prevent further damage to the core.

The Montague 1 and 2 design includes a radiation monitoring system with detectors on the four steam lines to detect, alarm, and isolate the steam lines if a gross fuel failure were to occur. We find this design is the same as on previously approved boiling water reactor designs and is acceptable.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

The principal components of the reactor coolant system are the reactor pressure vessel, the reactor recirculation system, the main steam and feedwater lines, and the pressure relief system. These items form the major components of the reactor coolant pressure boundary. The pressure boundary also contains portions of the reactor core isolation cooling system, the residual heat removal system and the reactor water cleanup system. Portions of these systems as well as other piping that extends from the reactor vessel out to the second outermost isolation valve are considered to be within the reactor coolant pressure boundary.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

The design loading combinations specified for ASME Code Class 1 components have been appropriately categorized with respect to the plant conditions identified as normal, upset, emergency or faulted. The design limits proposed by the applicants for these plant conditions are consistent with the provisions of Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." Conformance with the provisions of Regulatory Guide 1.48 for the design of the components will provide reasonable assurance that, in the event an earthquake should occur at the site, or of other system upset, emergency or faulted conditions, the resulting combined stresses imposed on the system components will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of General Design Criteria 1, 2 and 4.

5.2.1.1 Compliance with Code Requirements

We have reviewed the materials selection, toughness requirements and extent of materials testing proposed by the applicants to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary will have adequate toughness under test, normal and transient operation. All ferritic materials will meet the toughness requirements of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section III, as specified in

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10 CFR 50.55a. In addition, materials for the reactor vessel will satisfy the criteria of Appendix G, 10 CFR Part 50.

The fracture toughness tests and procedures required by the ASME Boiler and Pressure Vessel Code, Section III, as augmented by Appendix G, 10 CFR Part 50 for the reactor vessel provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant pressure boundary.

5.2.1.2 Compliance with 10 CFR Part 50, Section 50.55a

Components of the reactor coolant pressure boundary as defined by the rules of 10 CFR Part 50, Section 50.55a have been properly identified and classified as ASME Section III, Code Class I components in the application. These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and their addenda as specified by the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards. The staff concludes that construction of the components of the reactor coolant pressure boundary in conformance with the Commission's regulations provides reasonable assurance that the resulting quality standards are commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.1.3 Applicable Code Class

In Table 5.2.3 of the application, the applicants have identified the ASME Code Cases whose requirements will be applied in the construction of pressure-retaining ASME Section III, Code Class 1 components within the reactor coolant pressure boundary (Quality Group Classification A). The Code Cases in Table 5.2.3 are in accordance with those Code Cases in Regulatory Guides 1.84 and 1.85 and are acceptable to the NRC staff. The staff concludes that compliance with the requirements of these Code Cases in conformance with the Commission's regulations is expected to result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressure Protection

The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe operational transients and limits the reactor pressure during normal plant isolation and load rejections. The valves of the pressure relief system also are part of the automatic depressurization system, which is a subsystem of the emergency core cooling system, described in Section 6.3.

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The pressure relief system consists of 19 dual purpose safety/relief valves. All are mounted on the main steam lines within the primary containment drywell between the reactor vessel and the isolation valves. All safety/relief valves discharge through piping, directly to the suppression pool. The valves are all spring-loaded with the set pressures in the range from 1135 to 1205 pounds per square inch gage. At 103 percent of the set pressure of the valves, the valves have a combined capacity equal to 112 percent of rated steam flow. The valves are also actuated at relieving set pressures within the range of 1105 to 1145 pounds per square inch gage. These valves contain auxiliary pneumatic actuators and can be operated either by automatic or remote manual controls at any pressure above atmospheric. For overpressure relief, a pressure switch on each valve initiates the pneumatic actuator at the relieving set pressure. Eight of the valves can be pneumatically actuated by a signal from the automatic depressurization system. Each of these valves is equipped with a pneumatic accumulator and a check valve in the supply line so that the valve can be actuated even if the pneumatic supply fails.

The ability of the pressure relief system to prevent overpressurization of the reactor coolant pressure boundary is evaluated assuming that: (1) the plant is operating at design conditions (105 percent of rated steam flow and a reactor vessel dome pressure of 1045 pounds per square inch gage); (2) the most severe operational transient occurs (closure of the main steam line isolation valves); (3) the direct scram signal from the valve position switches fails and scram is effected by the fastest indirect scram signal (high neutron flux); and (4) one relief valve is inoperative.

An overpressure protection report which will present the results of analyses of overpressure transients such as turbine trip without bypass of steam to the condenser and closure of all main steam isolation valves will be submitted at the operating license review stage. This overpressure protection report will demonstrate that the design has met the acceptance criteria set forth in Section 5.2.2 of the NRC Standard Review Plan.

A small fraction of the pressure relief valves at some plants have inadvertently opened during certain transients. An evaluation of these inadvertent openings indicates that the potential exists for the same mechanism to prevent these valves from opening when required. Even though these failures have not resulted in overpressurization or compromised the integrity of the reactor containment system, they do represent a deviation from the anticipated performance of an essential safety system (overpressure relief system) that has safety implications such as excessive vessel cooldown rate, increased probability of coolant loss and potential for a common mechanism causing failure to open. Changes in design, equipment, inspection and testing can be made to improve the safety and safety/relief valves' performance.

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A new valve design will be used for Montague 1 and 2. The safety/relief valves to be used on these and similar plants will be balanced type, spring-loaded safety valves provided with an auxiliary power actuated device which allows opening of the valve even when pressure is less than the safety-set pressure of the valve. Valve problems on operating plants were associated principally with multiple stage pilot operated safety/relief valves. These newer, power operated safety valves employ significantly fewer moving parts wetted by the steam, and are therefore considered an improvement over the previously used valves.

Design details and drawings of the valves have been provided to the NRC staff. In addition, appropriate "bench" test data have been provided to verify improved performance. The General Electric Company intends to maintain a surveillance program once the new valves become operational on any plant. Based on our review of the new safety relief valve design and the surveillance program proposed by the General Electric Company, we conclude the safety relief valve design for Montague 1 and 2 is acceptable.

5.2.3 Materials Specifications and Compatibility with Reactor Coolant

The materials used for construction of components of the reactor coolant pressure boundary have been identified by specification, and found to be in conformance with the requirements of Section III of the ASME Code.

The materials of construction that will be exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as demonstrated by extensive testing and satisfactory performance. General corrosion of all materials except carbon and low alloy steel will be negligible. Conservative corrosion allowances have been provided for all exposed surfaces of carbon and low alloy steel in accordance with the requirements of the ASME Code, Section III.

Further protection against corrosion problems will be provided by control of the chemical environment and composition of the reactor coolant. The controls imposed on the fabrication of stainless steel and the anticipated results of demonstration tests in progress to show conformance to the provisions of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," provide reasonable assurance that components will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of structural integrity of a component. The materials of construction are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

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Conformance with the recommendations of the Regulatory Guide and the use of materials of proven performance constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria 1 and 14.

5.2.3.1 Fabrication and Processing of Ferritic Materials

Materials selection, toughness requirements, and extent of materials testing proposed by the applicants provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary and Code Class 2 and Class 3 components will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials are specified to meet the toughness requirements of the ASME Code, Section III and applicable NRC staff positions. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G, 10 CFR Part 50.

The fracture toughness tests and procedures required by Section III of the ASME Code, as augmented by Appendix G, 10 CFR Part 50, for the reactor vessel, and the staff position on Code Class 2 and Class 3 components provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components.

The results of the fracture toughness tests to be performed in accordance with the ASME Code and NRC regulations provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations and positions constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

The controls imposed on welding preheat temperature are in conformance with the provisions of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

The controls imposed on electroslag welding of ferritic steels are in accordance with the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties," and provide assurance that welds fabricated by this process will have high integrity and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance and postulated accident conditions.

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5.2.3.2 Fabrication and Processing of Austenitic Stainless Steel

The controls imposed upon components constructed of austenitic stainless steel conform to the provisions of Regulatory Guides 1.31, "Controls of Stainless Steel Welding" and 1.44, "Control of the Use of Sensitized Stainless Steel." The applicants have agreed to demonstrate the adequacy of current welding controls by conducting tests to determine the ferrite content of production welds and to evaluate the degree of sensitization in welded type 304 and 316 stainless steel. The examination of tubular products will be performed in accordance with the recommendations of Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissures) and in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with these Regulatory Guides constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

5.2.3.3 Mounting of Pressure-Relief Devices (Class 1)

The criteria used in developing the design and in mounting of ASME Class 1 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria used for the design and installation of ASME Class 1 overpressure relief devices constitute an acceptable design basis in satisfying the applicable requirements of NRC General Design Criteria 1, 2, 4, 14 and 15 and are consistent with those specified in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

5.2.3.4 Inservice Testing

To ensure that all ASME Code Class 1, 2 and 3 pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant, the applicants have agreed to conduct a test program which will include baseline preservice testing and periodic inservice testing. Such a program will provide for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress.

The applicants have stated that the inservice test program for all Code Class 1, 2 and 3 pumps and valves will meet the requirements of the ASME Code, Section XI, as defined in the proposed rules, Federal Register, Volume 39, No. 199 September 30, 1974. Specific details of the testing program will be evaluated during the operating license stage of our review.

Compliance with the referenced code requirements constitutes an acceptable basis for satisfying the applicable portions of General Design Criteria 37, 40, 43 and 46.

5.2.4 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones of Code Class 1 systems will be inspected periodically. The applicants have stated that the design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that suitable equipment is being considered to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection System

The leakage detection system proposed for leakage to the containment will include diverse leak detection methods, will have sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts. The major components of the system are the containment atmosphere radioactivity monitors, the containment sump monitoring system and the condensate flow monitoring system. Indirect indication of leakage will be obtained from the containment pressure and temperature indicators.

The design of the leakage detection system proposed to detect leakage from components and piping of the reactor coolant pressure boundary are in accordance with the provisions of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and provides reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions to be taken.

Compliance with the provisions of Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 30.

5.2.6 Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet the requirements of American Society for Testing Materials Standard E-185-73. This program also complies with Appendix H of 10 CFR Part 50 except that capsule holder brackets will be attached to the vessel cladding. We have reviewed the design fabrication and attachment of the capsule holder brackets. The design, analyses and fabrication of the capsule holder brackets are in accordance with the requirements of the ASME Code Section III and additional non-destructive testing will be conducted to ensure the integrity of the reactor vessel cladding and base metal in the areas where the brackets are attached. Based on the information provided concerning the design, fabrication and inspection for the installation of the capsule holder brackets we conclude that the attachment of the capsule holder brackets to the vessel cladding is acceptable and will not result in degradation of the reactor vessel base metal.

Changes in the fracture toughness of the material in the reactor vessel beltline caused by exposure to neutron irradiation will be assessed by a material surveillance program conforming to the requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50. Compliance with these requirements will ensure that the surveillance program constitutes an acceptable basis for monitoring irradiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of NRC General Design Criterion 31.

5.2.7 Operating Limitations

The plants will be operated in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR Part 50 to minimize any possibility of rapidly propagating failure. The use of Appendix G of the ASME Boiler and Pressure Vessel Code as a guide in establishing safe operating limits, using results of the fracture toughness tests performed in accordance with the Code and with NRC regulations, will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions. Compliance with these Code provisions and NRC regulations, provides an acceptable basis for satisfying the requirements of NRC General Design Criterion 31.

5.2.8 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude that there are no special considerations (Commission Memorandum and Order in the Matter of Consolidated Edison Company of New York, Indian Point Unit No. 2, Docket No. 50-247, October 26, 1972) that make it necessary to consider potential vessel failure.

The bases for our conclusion are that the design, material, fabrication, inspection and quality assurance requirements will conform to the rules of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section III.

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The stringent fracture toughness requirements of the ASME Code will be satisfied. Also, operating limits on temperature and pressure will be established in the technical specifications for this plant in accordance with the ASME Boiler and Pressure Vessel Code, Section III, and Appendix G, 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel:

(1) Will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases;

(2) Will be made from materials of controlled and demonstrated high quality;

(3) Will be subjected to extensive inspection and testing to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies;

(4) Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal operation or during upsets in operation, and that the reactor vessel will not fail under the conditions of any of the postulated accidents; and

(5) Will be subjected to monitoring and periodic inspection to determine that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

5.3 Thermal Hydraulic System Design

5.3.1 Analytical Methods and Data

The analytical methods, thermodynamic data and hydrodynamic data used are the same as those used in the GESSAR application and are acceptable to the NRC staff. These are also discussed in Section 4.4.

5.3.2 Load Following Characteristics

A boiling water reactor system as a result of its inherent load following characteristics, is able to follow load demands over a substantial range without requiring operator action. The power can be controlled over approximately a 25 percent power range by flow control. Because of the negative void coefficient, load following is accomplished by varying the reactor recirculation flow. To increase power, the recirculation flow rate is increased thus sweeping voids from the moderator and increasing core reactivity. As reactor power increases, more steam is formed and the reactor stabilizes at a new and higher power level with the transient excess reactivity balanced by the new void formation. Conversely, when less power is required the recirculation flow rate is reduced. The resultant formation of more

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voids in the moderator automatically decreases the reactor power to that commensurate with the new and reduced recirculation rate.

The transient effects of such events as full or partial loss of coolant flow, load changes, coolant pump speed changes, and startup of an inactive loop are discussed in Section 15.2 of this report.

5.4 Component and Subsystem Design

5.4.1 Reactor Recirculation System

The reactor recirculation system consists of two loops external to the reactor vessel, within the drywell, that provide for automatic load following capability over the range of 75 to 100 percent of rated power. The loops provide the piping path for the driving flow of water to the 20 reactor vessel jet pumps. Each loop contains one high capacity (constant speed) motor-driven pump, a flow control valve, two motor operated gate valves (for pump maintenance), and a bypass around the discharge gate and flow control valves. In each loop, subcooled water leaves the vessel in a 22-inch suction line and enters the suction of the recirculation pump.

The water is discharged at a head of 865 feet and a flow rate of 35,400 gallons per minute. The flow control valve normally varies the flow rate over a range from 75 to 100 percent power. The water from the recirculation pumps flows to 20 (10 per loop) jet pumps which are located in the reactor vessel and accelerates a portion of the flow in the annulus. The water that is in the annulus is returned to the recirculation pump through the suction lines. There are various system interlocks on the flow control valves and bypass valves to assure that adequate net positive suction head will be available and thus protect the pump from bearing or cavitation damage.

During their review of light water reactors, the Advisory Committee on Reactor Safeguards listed the potential for missiles resulting from recirculation pump motor overspeed as a generic concern requiring resolution satisfactory to the NRC staff. A decoupling device will be installed on the shaft between the pump and the motor such that in the event of a loss-of-coolant accident, the overspeed of the motor due to the "turbining" of the pump will not generate missiles which could cause the loss of any engineered safety feature. We have reviewed the design information and find it acceptable.

5.4.2 Main Steam Line Flow Restrictors

Each steam line is provided with a venturi-type flow restrictor within the drywell (between the reactor vessel and the first main steam line isolation valve). The restrictors limit flow to 200 percent of the rated flow, should a main steam line break occur outside the primary containment or downstream of the restrictors. The

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purpose of the restrictor is to limit the coolant blowdown loss prior to isolation valve closure to reduce (1) the probabilities and consequences of fuel failure, and (2) the forces on the reactor internal structure during blowdown. The restrictors are designed and fabricated in accordance with the ASME Code, Section III and are acceptable.

5.4.3 Main Steam Line Isolation Valves

Rapid acting isolation valves are located on each steam line on each side of the primary containment. On receipt of various signals from the plant protection system these valves close and isolate the reactor from other portions of the plant. At the same time isolation occurs, the same signals from the plant protection system are sent to various backup and emergency systems so that they automatically function as described in Section 6.3.

The analysis of a sudden, complete steam line break outside the containment is described in Chapter 15 and shows that the fuel clad is protected and the calculated accident doses are within 10 CFR Part 100 guidelines if the isolation valve closes in 5.5 seconds or less. We find this to be acceptable.

5.4.4 Reactor Core Isolation Cooling System

The reactor core isolation cooling system is a backup, high pressure source of reactor coolant that will operate independently of the normal alternating current power supply. Its operational purpose is to provide an alternate source of reactor coolant into the reactor vessel and to provide sufficient coolant to remove residual heat following reactor shutdown and loss of feedwater flow, without requiring depressurization of the reactor. The system consists of a pump driven by a steam turbine, taking steam from one of the main steam lines upstream of the isolation valves and adjacent to the reactor. The pump takes suction from either the condensate tank or the suppression pool and discharges it to the reactor vessel through a head spray nozzle. The system is designed to seismic Category I and Class I standards and is capable of being tested while the reactor is in operation. It has also been classified as an engineered safety feature but it is not part of the emergency core cooling system.

The reactor core isolation cooling system includes the piping, valves, pump, turbine, instrumentation, and controls used to maintain water inventory in the reactor vessel whenever it is isolated from the main feedwater system. The high pressure core spray system provides a redundant backup for this function. The scope of review of the system for Montague 1 and 2 includes the piping and instrumentation diagrams, equipment layout drawings, and functional specifications for essential components.

The drawings, component descriptions, design criteria, and supporting analysis have been reviewed and have been found to conform to applicable General Design Criteria and

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to appropriate Regulatory Guides and technical positions. The reactor core isolation cooling and the high pressure core spray system have been found capable of transferring core decay heat following a feedwater isolation and reactor shutdown, from the reactor to the suppression pool, so that the core minimum critical power ratio does not decrease below 1.07 and the pressure within the reactor coolant pressure boundary does not exceed 110 percent of design pressure. This capability has been found to be available even with a loss of offsite power and with a single active failure. The staff concludes that the proposed design of the reactor core isolation cooling system conforms to the Commission's regulations and to applicable Regulatory Guides and staff technical positions and is acceptable.

5.4.5 Residual Heat Removal System

The residual heat removal system is designed for two principal normal modes of operation besides the safety-related modes. For normal usage, the system functions to remove reactor decay and residual heat during either a normal shutdown or following isolation of the reactor. In one safety-related mode of operation, the system provides low pressure coolant injection to restore and maintain the coolant inventory in the reactor vessel after a postulated loss-of-coolant accident. In the other safety-related mode of operation, the system provides a containment spray for condensing steam in the containment during the post-accident period. These safety-related modes of operation of the system are further discussed in Section 6.3 of this report.

The system consists of two heat exchangers, three main system pumps, and associated valves, piping, controls and instrumentation. The main system pumps are sized on the basis of flow required during the low pressureinjection mode of operation, which is the mode requiring the maximum flow rate. The heat exchangers are sized on the basis of their heat removal duty following a loss-of-coolant accident.

Two loops, each consisting of one heat exchanger and one pump and auxiliary equipment, are physically separated from each other in the reactor building. A third loop, also consisting of a pump and associated piping, can pump service water directly into the reactor, if necessary.

During reactor isolation, the residual heat removal system can be operated in the condensing mode to condense reactor steam; hence, the system can also operate in conjunction with the reactor core isolation cooling system. With the reactor isolated, reactor steam normally is directed to and condensed in the suppression pool via the relief valves and the reactor isolation cooling turbine exhaust piping. However, the suppression pool temperature under these conditions is limited so that in the event of a postulated design basis loss-of-coolant accident the water temperature rise would not cause the pool temperature to exceed 170 degrees Fahrenheit during the reactor blowdown. The condensing mode of operation relieves the burden on the suppression pool by transferring a portion of the steam generated by decay heat to the

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service water. The condensate is either dumped to the suppression pool or returned to the reactor vessel through the suction of the steam-turbine driven pump. Shortly after shutdown, both heat exchangers are used to handle essentially all of the decay heat. After about 1-1/2 hours, the capacity of one heat exchanger is adequate and the other may be transferred to the suppression pool cooling mode, which utilizes the heat exchanger to cool the suppression pool water by transferring heat to the service water system. This mode can be used in conjunction with the condensing mode or to provide for long-term suppression pool cooling following a postulated loss-of-coolant accident.

The shutdown cooling mode and a reactor vessel head spray mode are operated during normal shutdown and cooldown. Reactor water is diverted from one of the recirculation loops, through the system pumps and heat exchangers (shell side) where heat is transferred to the residual heat removal service water (tube side), then the cooler reactor water is returned to the reactor vessel via one or both recirculation loops. Part of the cooled reactor water flow is diverted to a reactor head spray nozzle where it maintains saturated conditions in the vessel head volume by condensing the steam generated by the hot vessel walls and internals. The system is protected against overpressurization by relief valves and can be automatically isolated to protect the core from low water level in case of a break in the system piping.

The scope of our review of the residual heat removal system included the piping and instrumentation diagrams, equipment layout drawings, and performance specifications for essential components. Our review also included the applicants' proposed design criteria and design bases, and their analysis of the adequacy of those criteria and bases and how well the design conforms to these criteria and bases.

Based on our review of the drawings, component descriptions and design criteria, we find that the residual heat removal system is similar to those previously reviewed by the staff. The system does not meet the intent of General Design Criteria 19 and 34 which requires shutdown from the control room with the assumed most restrictive single active failure. The applicants have proposed alternate procedures for providing shutdown cooling in the event that the normal residual heat removal system does not function. These procedures and other proposals are being reviewed by the staff on a generic basis for the BWR-6 class of plants. The applicants have agreed to incorporate the generic resolution of this issue, which will be developed during our continuing review of the GESSAR application, into their design. We will require that this matter be resolved to our satisfaction prior to a decision on issuance of construction permits and will describe its resolution in a supplement to this report.

5.4.6 Reactor Water Cleanup System

5.4.6.1 System Description and Evaluation

The reactor water cleanup system will include a separate system for each reactor. Each system will consist of two mixed bed filter/demineralizers (Powdex), each having a design capacity of 77,000 pounds per hour, which will be run in parallel.

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Two isolation valves in the system will close automatically on a signal from the reactor coolant pressure boundary leak detection system to prevent the loss of coolant and the release of radioactive material from the reactor vessel. These valves will also operate if the standby liquid control system is activated or if the outlet temperature of the nonregenerative heat exchangers exceeds a preset level.

The design of the valves is such that they can be operated manually. Reverse flow isolation will be provided by at least one check valve in the system or feedwater piping.

Flow will be maintained in the filter/demineralizers in the event of low flow or loss of flow by separate holding pumps provided for each filter/demineralizer unit. Resin loss will be prevented by strainers on the outlet of each filter/demineralizer unit.

Those system components which will be within the outermost isolation valve boundary will be designed to Quality Group A and seismic Category I requirements. Those components outboard of the outermost isolation valve boundary will be designed to Quality Group C and non-seismic requirements. The isolation valves will be designed to Quality Group A and seismic Category I requirements.

5.4.6.2 Evaluation Findings

The reactor water cleanup system will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The scope of our review of the system included the system capability to meet the anticipated needs of the plant, the capability of the instrumentation and process controls to ensure operation within the recommended limits defined in Regulatory Guide 1.56-Maintenance of Water Purity in Boiling Water Reactors, and the seismic design and quality group classifications relative to Regulatory Guides 1.26 and 1.29. Our review included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

The basis for acceptance in our review has been the conformance of the applicants' design and design criteria to the Commission's regulations and to applicable regulatory guides, as referenced above, and to applicable staff technical positions and industry standards.

Based on the foregoing evaluation, we conclude that the proposed design for the reactor water cleanup system is acceptable.

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6.0 ENGINEERED SAFETY FEATURES

6.1 General

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials, should a major accident occur in the plant. In this section of our report, we discuss the reactor containment system, the emergency core cooling systems, and the provisions for maintaining the habitability of the control room after postulated accidents. Discussions of other engineered safety features are provided elsewhere in this report, as related to the particular systems they directly serve. As will be seen, certain of these systems have functions for normal plant operations as well as safety-related functions.

Systems and components designated as engineered safety features are designed to be capable of performing their function of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They are designed to seismic Category I standards and they must function even with a complete loss of offsite power. Components and systems are provided with sufficient redundancy so that a single failure of any component or system, will not result in the loss of the plant's capability to achieve and maintain a safe shutdown of the reactor. The instrumentation systems and emergency power systems are designed to the same seismic, redundancy, and quality requirements as the systems they serve. These instrumentation and onsite power systems are described in Sections 7 and 8, respectively, of this report.

6.1.1 Engineered Safety Feature Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold worked stainless steel shall be taken to be less than 90,000 pounds per square inch.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the provisions of Regulatory Guides 1.31, "Control of Stainless Steel Welding" and 1.44, "Control of the Use of Sensitized Stainless Steel." The applicants have agreed to demonstrate the adequacy of welding procedures by conducting tests to determine the ferrite content of production welds and to evaluate the degree of sensitization in welded type 304 and 316 stainless steel. Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress corrosion cracking will not occur. The controls placed on the concentration of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel componenets are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The applicants have stated and we agree that the absence of chemical sprays eliminates the necessity of considering deleterious corrosion of structural elements located inside the containment systems or the generation of hydrogen gas from corrosion reactions.

Conformance with the Codes, Regulatory Guide recommendations and positions mentioned above constitute an acceptable basis for satisfying the requirements of General Design Criteria 35, 38 and 41.

6.2 Containment Systems

The containment systems for each of the units include a reactor containment structure, containment heat removal systems, containment isolation system, combustible gas control system, a shield building surrounding the primary containment and a standby gas treatment system. The design of the containment systems for Montague 1 and 2 is referred to as the Mark III design. Other recently reviewed plants have utilized the Mark III design, the first being the Grand Gulf Nuclear Station, which is now under construction. The GESSAR-238 Nuclear Island Standard Design also incorporates the Mark III design as do the Allens Creek, Clinton, Perry and River Bend plants. The design for Montague 1 and 2 is particularly related to River Bend, as both designs were provided by the Stone & Webster Engineering Corporation. A comparison of the principal containment parameters for Montague 1 and 2, River Bend and GESSAR are summarized in Table 6.2.1.

Issues raised with respect to the Montague 1 and 2 design included generic items originally discussed during our reviews of the previous plants with Mark III containments. Grand Gulf was the first plant to be reviewed by the staff and during its review, the basic analytical approach and design margins were established and the scope of the large-scale Mark III test prog as determined. Based on successful resolution of these issues, the G ulf de ign was found to be acceptable pending final validation of the analytical model with large-scale test data. The River Bend analysis was similar in concept, but used an analytical model developed by Stone & Webster. This approach, for example, includes consideration of containment heat sinks which were not included in previously reviewed Mark III designs; however, we found the use of heat sinks acceptable as a result of our detailed review. As discussed in the River Bend Safety Evaluation Report, the Stone & Webster analysis was found to be acceptable with the inclusion of a 30 percent margin on drywell pressure. A similar approach has been taken for Montague 1 and 2. In addition, the staff now has available in the CONTEMPT computer code, the capability for calculating the pressure-temperature response of a Mark III containment. The results of our independent calculations were used to confirm the applicants' analysis.

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TABLE 6.2.1

DESIGN PARAMETERS

	River Bend	GESSAR	Montague
Drywell Volume, cubic feet	247,000	274,500	274,600
Drywell Construction	Reinforced concrete	Reinforced concrete	Reinforced concrete
Containment Volume, cubic feet	1,120,000	1,168,000	1,375,000
Containment Construction	Steel shell	Steel shell	Steel shell
Drywell Design Pressure, pounds per square inch difference	25	30	26
Containment Design Pressure, pounds per square inch gage	15	15	15
Reactor Power, megawatts thermal	3015	3758	3758
Number of Vents	129	120	141
Vent Diameter, inches	25.25	27.5	27.5
Total Vent Area, square feet	449	480	569
Suppression Pool Volume, cubic feet	126,600	163,700*	159,000
Break Area (DBA), square feet	3.275	3.94	3.94
Break Area/Vent Area	.0073	.0082	.0069
Initial Blowdown Mass Release, pounds mass	506,600	609,000	609,000
Initial Blowdown Energy Release, British thermal units	328×10^6	334×10^6	334×10^6
Vent submergence, feet	7.5	7.5	6.5

* With upper pool dump.

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6.2.1 Containment Functional Design

The containment system is divided into two major subvolumes, a drywell enclosing the reactor system, and the primary containment surrounding the drywell and containing the suppression pool. The containment and the drywell volumes are connected through the suppression pool by an array of horizontal vents in the drywell wall. The suppression pool serves as a large heat sink in the unlikely event of a design basis loss-of-coolant accident to condense steam released from the reactor plant.

The primary containment is a free-standing steel structure consisting of a vertical cylinder, domed top, and a flat base. The net free volume of the primary containment is 1.375×10^6 cubic feet and the design pressure is 15 pounds per square inch gage. To satisfy its design basis as a fission product leakage barrier, the primary containment is designed for a leak rate of 0.275 percent of the volume per day at 8.52 pounds per square inch gage.

A low pressure structure called the shield building, surrounds the primary containment. Its purpose is to provide an enclosed volume in which most of the fission product leakage from the primary containment following a postulated loss-of-coolant accident can be held up and filtered prior to release to the environment. Our evaluation of the shield building design is included in Section 6.2.3 of this report.

Located within the primary containment is a substructure, called the drywell, which encloses the reactor and reactor coolant system. The drywell is an unlined concrete structure, enclosing a net free volume of 274,600 cubic feet and designed for a differential pressure of 26 pounds per square inch difference. The purpose of the drywell is to channel the steam released from the reactor system during an unlikely loss-of-coolant accident through a horizontal vent system matrix to the suppression pool for condensation. While not a fission product barrier, the drywell must be free of gross leakage for adequate performance of the pressure suppression feature to assure proper channeling of the steam to the suppression pool.

Since, for the Mark III design, the containment completely surrounds the drywell, high energy lines penetrating the drywell must pass through the containment and shield building volumes. These lines are designed to low stress levels and high quality standards to minimize the probability for rupture inside the containment but outside the drywell. As an additional margin, the applicants have provided guardpipes on high energy lines extending from the drywell through the containment to the shield building. The guardpipes will be designed to the pressure of the enclosed process line.

Because the pressure suppression concept relies upon a controlled channeling of steam through the suppression system, the possibility of bypass paths must be minimized. Our evaluation of potential bypass sources and containment bypass capability is discussed in Section 6.2.1.4 of this report.

The suppression pool is a 360-degree annular pool located in the bottom of the containment and retained between the containment wall and the drywell weir wall. The weir wall is a 360-degree reinforced concrete wall located inside the drywell and approximately 33 inches from the drywell wall. At a normal water height of 18 feet, the volume of water in the suppression pool is 159,000 cubic feet. The suppression pool serves both as a heat sink for postulated transients and accidents and as the source of cooling water for the emergency core cooling systems. In the case of operating transients that result in a loss of the main heat sink, energy would be transferred to the pool by the discharge piping from the reactor pressure relief or safety valves. In the event of a loss-of-coolant accident within the drywell, the horizontal vent system in the drywell wall would provide the energy transfer path.

Located in the vertical section of the drywell wall and below the suppression pool water level are 141 vent holes of 27.5 inches in diameter and arranged in 47 circumferential columns of three vents. In the event of a loss-of-coolant accident the pressure will rise in the drywell due to the release of reactor coolant, and force the level of water down in the weir annulus. When the water level has been depressed to the level of the first row of vents, the differential pressure will cause air, steam and entrained water to flow from the drywell into the suppression pool. The steam will be condensed in the pool and the air driven from the drywell will be compressed in the primary containment. The net effect could result in approximately a 4 pound per square inch rise in average containment pressure. Peak drywell differential pressure is calculated by the applicants to be 19.6 pounds per square inch difference.

Figure 6.2.1 illustrates the drywell and containment pressure response as a function of time following the design basis loss-of-coolant accident. In Figure 6.2.1, the short term containment response is shown in terms of two regions; one representing the volume between the suppression pool and the hydraulic control units floor, and the other representing the remainder of the containment volume. This two-node modeling of the containment is discussed further in Section 6.2.1.1.

Following the initial phase of the accident, containment and drywell pressure will continue to rise due to the input of core decay and sensible heat to the suppression pool. The long-term pressure rise will be limited to 8.52 pounds per square inch gage by operation of any one of the redundant containment heat removal systems as well as the effect of the containment passive heat sinks. Therefore, in the pressure response analysis of this type of containment, two limiting conditions must be considered; the short-term drywell differential pressure and the long-term containment shell pressure. Our evaluation of the applicants' analytical methods for each of these time periods (i.e., both long and short term) is discussed in Sections 6.2.1.1 and 6.2.1.2 of this report. The General Electric Company has also completed small-scale tests and is performing large-scale tests to support the Stone & Webster Mark III short-term analytical model. Our review of these test programs is also discussed below.

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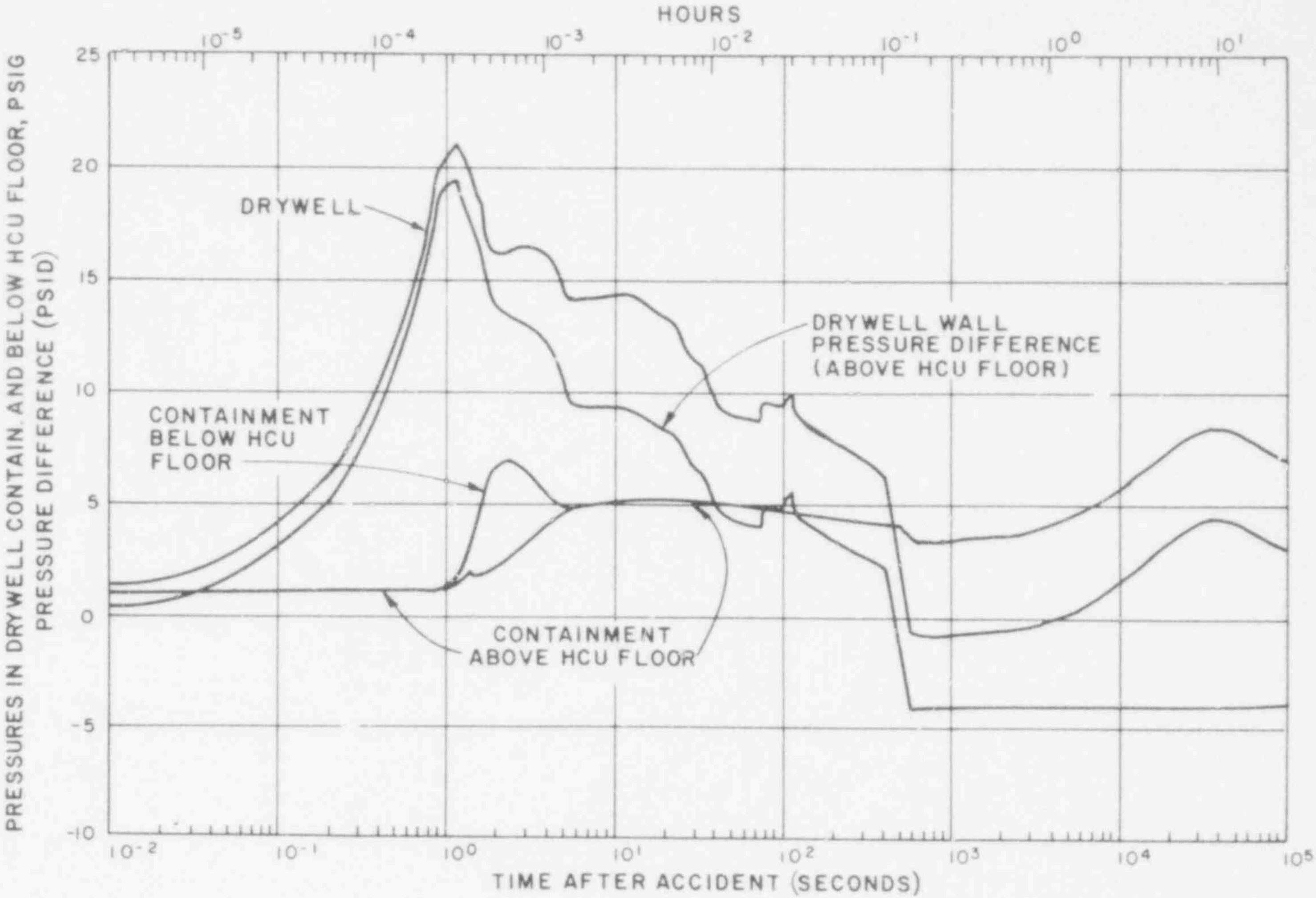

HCU = Hydraulic Control Units

Figure 6.2.1 – PRESSURE RESPONSE FOR DRYWELL AND CONTAINMENT

6.2.1.1 Short-Term Pressure Response

For containment analysis, a modified LOCTVS computer program, as described in "LOCTVS - A Computer Code to Determine the Pressure and Temperature Response of Pressure Suppression Containments to a Loss-of-Coolant Accident," SWD-2 and supplements, was used. The program performs numerical integrations principally of the mass and energy conservation equations, and also of the momentum conservation equation as required to determine flow rates between nodes. LOCTVS was written by Stone & Webster Engineering Corporation to simulate behavior of the pressure suppression containment system, the reactor coolant system, the containment heat sources and sinks, and the containment heat removal system. The program has been modified to allow computation of the unique vent clearing phenomenon associated with the Mark III concept. The analytical approach used for these modifications was submitted as part of the River Bend Station, application Docket Nos. 50-458/459, and will be further documented in a future topical report by Stone and Webster.

The model of the vent system is a six-control volume nodalization representing the vertical weir annulus and horizontal vents. Conservation of mass and momentum is applied to each control volume to determine fluid accelerations and vent clearing times. An additional node is used to simulate the suppression pool. Turning loss coefficients are applied in a manner similar to the General Electric approach to account for changes in flow path direction and area. The loss coefficients currently used in the model are derived from generally accepted data. Stone & Webster intends to verify these coefficients during the large-scale Mark III testing program.

In addition to the calculation of vent dynamics, LOCTVS also contains the capability for blowdown calculations. The primary system is modeled as a single node with the water inventory considered as a pure steam volume and a pure liquid volume creating a distinct interface between phases. For the postulated rupture, the frictionless Moody critical flow model is used to compute the blowdown rate.

Using the LOCTVS program, the applicants have calculated the short-term drywell pressure response for both a postulated main steam line break and recirculation line break. Similar to previously reviewed Mark III designs, the postulated main steam line rupture was found to be the limiting break.

For the postulated double-ended rupture of the main steam line, the blowdown is divided into two distinct phases. Initially the blowdown is pure steam. However, as the blowdown begins to depressurize the system, the liquid level in the reactor vessel swells to the elevation of the steam nozzles. At this time, the blowdown changes to two-phase due to the entrained liquid. Peak drywell differential pressure can be sensitive to the level rise time since two-phase blowdown yields a greater rate of steam addition to the drywell than steam only blowdown and also introduces liquid water into the vent flow. Both of these effects increase drywell pressure.

For the Montague containment analysis, Stone & Webster has computed a level rise time of about one second. Based on this calculation, the peak drywell differential pressure is computed to be 19.6 pounds per square inch difference.

Following the postulated design basis loss-of-coolant accident, the drywell pressure will rise and accelerate the water in the vent annulus. At about 0.9 second, the first row of vents will be cleared of water and a mixture of air, steam and water will flow into the suppression pool. The water will continue to accelerate downward resulting in clearing of the second row at about 1.17 seconds and the third row at about 1.64 seconds. The peak drywell differential pressure occurs at 1.17 seconds (main steam line break) and is a result of sufficient vent area being uncovered to reverse the pressure transient.

The applicants have stated that the drywell will be designed for an internal pressure of 26 pounds per square inch difference which provides a margin of 32 percent above the peak calculated value. Both the NRC staff and its consultant, the Aerojet Nuclear Company, have reviewed the analytical model used for the drywell pressure response calculation. We have also performed our own calculations of the drywell pressure response using the CONTEMPT-LT computer code. Our results confirm the validity of the applicants' analysis. Based on this confirmation, our review of the applicants' analytical model, and our consultant's recommendations, we conclude that the proposed drywell design pressure is acceptable.

As shown in Figure 6.2.1 the short term containment response is calculated for two regions; the volume between the suppression pool and hydraulic control unit floor, and the remainder of the containment volume. This represents a recent modification to the LOCTVS code which was required to describe the flow restrictive effect of the floor. For Montague 1 and 2, this floor blocks 50 percent of the annulus area between the drywell and containment walls and could restrict the flow of non-condensibles during the initial blowdown phase. The applicants calculate a containment pressure rise of approximately 7.5 pounds per square inch below the floor. At this time, however, the applicants have not submitted the details on the revised LOCTVS model, or justification that may be appropriate in terms of relevant test data from the large-scale Mark III test facility. We will review this information at the operating license stage of review since the applicants have committed to increasing the available flow area at the floor if required to maintain a 30 percent pressure margin for the drywell design pressure. We conclude that this commitment to maintain the 30 percent pressure margin for the drywell design is acceptable for the construction permit stage of our review.

6.2.1.2 Long-Term Pressure Response

Following the short-term blowdown phase of the accident, suppression pool temperature and containment pressure will increase due to the continued input of decay and sensible heat into the containment. During this time period the emergency core cooling system pumps, taking suction from the suppression pool, will have reflooded the reactor pressure vessel up to the level of the main steam line nozzles. Subsequently emergency core cooling system water will overflow out the break and fill the drywell up to the top of the weir wall, establishing a recirculation flow path for the coolant.

After 10 minutes following the accident, the containment cooling mode of the residual heat removal system is activated and suppression pool water is circulated through the systems heat exchangers, establishing an active energy transfer path to the service water system and ultimate heat sink. In addition to the residual heat removal system, passive heat sinks within the containment were considered as a heat rejection mechanism. This is similar to the method of analysis used for River Bend.

For the long-term analysis, the applicants have conservatively accounted for potential post-accident energy sources. These include decay heat, sensible heat, pump heat, and metal-water reaction energy.

Based on the above assumptions, the applicants have calculated the peak containment pressure to be 8.5 pounds per square inch gage. The design pressure of the containment is 15 pounds per square inch gage which allows a 76 percent margin above the peak calculated value. On the basis of our review of the applicants' analysis and the pressure margin, we conclude that the containment design pressure for this plant is acceptable.

The drywell structure is designed for an external pressure of 20 pounds per square inch difference. We have reviewed the drywell design external pressure and find that it is acceptable since it represents an upper limit on possible external pressures by assuming complete depressurization of the drywell with the containment at its post-blowdown pressure of about 20 pounds per square inch atmosphere.

6.2.1.3 Subcompartment Pressure Analysis

Within both the drywell and containment, internal structures form subcompartments or restricted volumes which are subject to differential pressures following postulated pipe ruptures. In the drywell there are two such volumes; the annulus formed by the reactor vessel and the biological shield, and the drywell head region which is a cavity surrounding the reactor pressure vessel head. In the containment the various components of the reactor water cleanup system are located in individual compartments.

The applicants have submitted the results of calculations of pressure differentials across the walls of these subcompartments. We have performed similar analyses for the drywell head region and reactor water cleanup system compartments. Our calculations confirm the applicants' results with the exception of the drywell head. We believe that our calculated results represent reasonable estimates of pressure differential across walls of compartments based on current calculational techniques. In its letter dated October 14, 1975 the applicants provided additional analysis of the drywell head subcompartment peak pressure difference based on an increased vent from 11 to 13.1 square feet. We have reviewed this calculation and find it, with the increased vent area, to be acceptable. The applicants have provided a description of modeling and methods to be used to obtain the pressure loads imposed on the reactor pressure vessel and the biological shield wall for a pipe rupture within the vessel shield annulus. We find the modeling and methods to be used acceptable. However, we will require the applicants to provide, for our review, the final results and loads to be used for the design of the vessel supports and shield wall prior to completing the construction permit stage of our review. The final design loads will be reported in a supplement to this report.

6.2.1.4 Steam Bypass of the Suppression Pool

There are three potential sources of steam bypass of the suppression pool associated with the Mark III containment design used for the Montague Nuclear Power Station. First, since the drywell is of reinforced concrete construction, the potential exists for cracking of the drywell structure under accident loading conditions. This will allow direct leakage of blowdown steam to the containment volume. Second, the design of the combustible gas control systems could allow the opening of direct flow paths between the drywell and containment for the dilution of hydrogen. Although these systems are designed to operate after blowdown is complete, residual steaming in the reactor vessel will continue after blowdown due to the addition of decay and sensible heat to the coolant. This energy could be added directly to the containment atmosphere. Thirdly, the reactor water cleanup system is located within the primary containment but outside the drywell. This system has high energy pipe lines, connected to the reactor primary system, which will not have guardpipes. Therefore, postulated ruptures in these lines would result in blowdown of reactor coolant directly to the containment atmosphere without benefit of energy absorption in the suppression pool.

In the case of postulated reactor water cleanup system pipe breaks, the applicants have provided design features to terminate the blowdown prior to exceeding the design limits of the containment. Isolation valves are provided on the system suction line which will automatically isolate the system from the primary reactor system. In addition, a flow limiter is provided in the suction line to limit the rate of blowdown prior to isolation. Backflow from the feedwater line is prevented by redundant check valves in the reactor water cleanup return line. The applicants have calculated that the containment pressure response assuming a reactor water cleanup pipe rupture would be less than one pound per square inch gage, which is below the containment design pressure of 15 pounds per square inch gage.

As described in Section 6.2.5, operation of the combustible gas control system would result in a direct flow path between the drywell and the containment. Although inadvertent opening of a single mixing system line is within the bypass

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capability of the containment, actual operation of the whole system should be prevented until the rate of bypassing is within acceptable limits. The applicants have provided appropriate interlocks for the mixing system that will prevent inadvertent actuation of the hydrogen mixing system for 10 minutes after an accident. This interlock is a timer which will prevent actuation of the mixing system for 10 minutes following an accident. Our review of the break spectrum effects as a function of time indicates that for small breaks the drywell pressure may exceed the containment pressure for longer than 10 minutes. The applicants have reduced the size of the mixing system piping to six inch diameter piping so that inadvertent operation of the mixing system will not exceed the allowable bypass capability of one square foot and satisfies this requirement for the small break as well as the large break. We conclude this design change is acceptable.

Possible bypass leakage paths from the drywell to the outer containment have been considered in our review of the Mark III containment. The control of such bypass paths is important to ensure that the design pressure of the containment is not exceeded for postulated design basis accidents. In regard to bypass leakage that could be associated with potential cracking of the drywell or other sources around penetrations, we believe that containments of the Mark III design should have an allowable bypass capability of about one square foot ($A/\sqrt{K}$) for the spectrum of reactor coolant system breaks. The term ($A/\sqrt{K}$) is the area of the bypass divided by the square root of the pressure drop loss coefficient. The allowable bypass area is considered to be that leakage area between the drywell and containment which would result in containment pressurization to design pressure following a postulated loss-of-coolant accident. The applicants have proposed a design which provides a bypass capability of about one square foot ($A/\sqrt{K}$). The capability is based on a 200 degrees Fahrenheit per hour reactor cooldown rate, operation of one containment cooler, and consideration of the available containment heat sinks. The selected cooldown rate is initiated if the containment pressure exceeds five pounds per square inch gage at 10 minutes after a postulated accident.

Two air coolers powered from separate diesels, will be used to aid in the condensation of bypass steam. The system will be automatically activated by loss of offsite power, drywell high pressure, or reactor low water level signals.

The applicants have made a commitment to perform a leakage test of the drywell at design pressure prior to plant operation, and low pressure leakage tests of the drywell periodically during plant lifetime. The applicants have agreed to establish the acceptance criterion for the tests based on the measured leakage being less than the leakage corresponding to a flow path of about 10 percent of the allowable $A/\sqrt{K}$ at the test pressure. We find this commitment acceptable for the construction permit stage of our review.

6.2.1.5 Test Program

The General Electric Company is presently conducting a large-scale test program to verify the performance characteristics of the Mark III containment. Large-scale testing was started in November 1973 following completion of a two-year small-scale test program.

A total of 67 small-scale tests have been performed since June 1971. The test arrangement simulates a Mark III containment with a volumetric scale of approximately 1:2000. Small-scale test data have been reported in "Mark III Confirmatory Test Program Progress Report," NEDM-10848 and "Mark III Analytical Investigations of Small Scale Tests Progress Report," NEDM-10976. The intent of these tests was basically proof of principle of a horizontal vent system and also a preliminary checkout of the vent clearing model. Correlations between test data and analytical predictions for vent clearing times indicated reasonable agreement in this scale.

The large-scale test program utilizes a facility which simulates a segment of a Mark III containment. The nominal volumetric scale factor of the facility is 1/130 with the exception of the vent system and suppression pool. Vent system test sections in full, one-third, and one-ninth scale are used with correspondingly scaled pool sections in the various stages of the test program. The original character of the program was to be a confirmatory exercise to verify the short-term analytical model described in Section 6.2.1.1. The scope of the program included testing beyond design basis conditions to investigate the margins available in pressure suppression systems. Additional "phenomena" tests are also planned (i.e., vent interaction) to confirm that their effect had been adequately treated in the analytical modeling.

A derivative of early tests, however, was the observation that containment structures could be subject to significant suppression pool hydrodynamic loads during blowdown (see Section 6.2.1.6). This has resulted in several additional tests whose objective was to generate design basis loads to be incorporated in the design of the affected containment structures.

Eleven large-scale test series have been completed to date. Discussions of these and future test series are provided below. A list of completed tests is provided in Table 6.2.2.

(1) Series 5701 - 5703

The primary objective of these tests was to verify short-term analytical models for horizontal vents. Tests were run with one, two, and three vents open (unplugged) for three scaled break areas (50, 100 and 20 percent of the design bases accident break) and centerline submergences of two to 12 feet. Stone & Webster is currently checking their modes against the measured test results.

(2) Series 5705 & 5706

Eleven air blowdown tests were performed using the full-scale (27-1/2 inches) test section with one of the three vents plugged. Impact targets were located above the test facility pool. Tests were run with submergences of six to 10

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TABLE 6.2.2

LARGE SCALE TESTS COMPLETED

Test Series	Blowdown	Vent Scale	No. of Tests	Primary Objective	Documentation
5701	Steam	Full	21	Vent Clearing	NEDM-13377
5702	Steam	Full	17	Vent Clearing	NEDO-20345
5703	Steam	Full	3	Vent Clearing	NEDO-20533
5705	Air	Full	4	Pool Swell	NEDO-20550
5706	Air	Full	7	Pool Swell	NEDE-20732P
5801	Steam	1/3	19	Pool Swell	NEDM-13407P
5802	Steam	1/3	3	Pool Swell	NEDM-13407P
5803	Water	1/3	2	Pool Swell	NEDM-13407P
5804	Steam	1/3	5	Pool Swell	NEDM-13407P
5805	Steam	1/3	51	Impact Loads	NEDE-13426P
5806	Air	1/3	12	Pool Swell	NEDE-13435P

feet and pool surface to target clearances of four to 18-1/2 feet. The objectives of the tests were to obtain scoping data regarding pool dynamic response and impact loads on structures located above the suppression pool. Air blowdown tests were required to achieve air charging rates into the pool which were representative of an actual plant. We find that the test results provided an early indication of the range of pool swell and the magnitude of impact loads on small structures.

(3) Series 5801 - 5804

Twenty-nine tests were run using the one-third scale vent test section (vent area scaled) with a one-third scale suppression pool (pool area scaled). The flow restriction at the hydraulic control unit floor was also modeled. The objectives of these tests series were to measure froth impingement loads on the floor and two-phase pressure drop across the floor, and to determine pool swell motion characteristics. Our review of this test data is currently in progress.

(4) Series 5805

This test series utilized the same facility arrangement as Series 5801 - 5804 and included pipes, I-beams and grating situated above the pool. The objective of this series was to measure pool impact loads on representative containment structures. The results of these tests were recently submitted to the NRC and are currently under review.

(5) Series 5806

Twelve air blowdowns were run in this test series utilizing the same facility arrangement as Series 5801-5804. The objectives of this series were to determine pool motion characteristics for large air mass fraction vent flows and to compare these one-third scale results with the previous full-scale air test results. The results of these tests were recently submitted to the NRC and are currently under review.

Integration of the pool dynamics test results into the Montague containment design is discussed in Section 6.2.1.6. Additional large-scale tests are planned as discussed below:

(1) A series of liquid blowdown tests will be conducted to indicate comparability to steam blowdowns.

(2) A series of small break tests will be conducted to investigate pool stratification and vent chugging effects.

(3) Tests will be performed with the suppression pool at an initial elevated temperature to determine steam condensation characteristics under such conditions.

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(4) A multi-vent series will be run employing a test section of three columns of three, nine-inch vents (one-ninth scale by area) to consider possible vent interactions.

As discussed in Sections 6.2.1.1 and 6.2.1.2, we consider the basic design and performance of the Mark III containment system to be well established based on our review of the analytical models and the available margins incorporated in the design. Pool dynamic loads are a localized phenomenon which have received additional consideration as discussed in Section 6.2.1.6.

In several recent reports concerning BWR plants with Mark III containments (e.g., Perry and River Bend, the Advisory Committee on Reactor Safeguards commented on the progress of the confirmatory test program. In particular, the Committee emphasized the importance of developing analytical models based on a first principles approach which can be used in conjunction with empirical test results. The applicants have indicated that the resolution of this issue on River Bend will be equally applicable to Montague 1 and 2 since Stone & Webster is performing the containment analyses for both plants. We are continuing our review of this matter with Stone & Webster on a generic basis. We will review this matter as it affects Montague 1 and 2 during the operating license stage of our review.

In summary, we consider the remaining Mark III testing to be confirmatory in nature; we will require that the tests and our evaluation of the test results be completed prior to issuance of the first operating license for a Mark III plant.

6.2.1.6 Pool Dynamics

Several phenomena have been identified in our review of the Mark III containment that could result in dynamic loading of structures located in and above the suppression pool. They are related to (1) pool response to the postulated loss-of-coolant accident and (2) pool response due to relief valve operation, generally associated with plant transient conditions. These phenomena are described in more detail below.

Accident Pool Dynamics: Following a postulated loss-of-coolant accident in the drywell, the drywell atmosphere will be compressed due to blowdown mass and energy addition to the volume. Following vent clearing an air/steam/water mixture will be forced from the drywell through the vent system and injected into the suppression pool, approximately seven to ten feet below the surface. The steam component of the flow mixture will condense in the pool, while the air will be released in the pool as high pressure bubbles. The continued addition and expansion of air causes the pool volume to swell resulting in an acceleration of the surface vertically upward. Due to the effect of buoyancy, air bubbles will rise faster than the pool water mass and

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will eventually break through the swollen surface and relieve the driving force behind the pool. Due to the dynamics of vent clearing and vent flow and the vertical motion of the pool water mass, structures forming the suppression pool boundary, structures located within the pool, and structures located above the pool could be subject to hydrodynamic loads.

Relief Valve Dynamics: Pressure waves are generated within the suppression pool when, on first opening, relief valves discharge high pressure air and steam into the pool water. This phenomenon is referred to as relief valve vent clearing loads which are imparted to pool retaining structures and structures located within the pool. These same structures can also be subject to loads which accompany extended relief valve discharge into the pool if the pool water is at an elevated temperature. This effect is known as steam quenching vibrations.

A letter was sent to the applicants dated April 22, 1975, which describes potential loads due to pool dynamics and which requests a description of the manner by which these loads were considered in the containment design. With regard to relief valve dynamics the applicants have submitted information describing vent clearing loads based on a Stone & Webster analytical model. We will require the applicants to submit the remainder of the requested information on relief valves including a description of the analytical model and its experimental justification. With regard to accident pool dynamics the applicants have made general reference to the GESSAR docket for resolution of our concerns with the qualification that significant variations in the Montague design from that of GESSAR will be addressed separately.

We have concluded that in some instances the design loads were inadequately substantiated by test data or were based on what the NRC staff considered to be a non-conservative interpretation of the test data. We based this on our review of the information given in General Electric quarterly progress reports issued through April 1975 for the Mark III Confirmatory Test Program. Accordingly, in order to assure that the results of the ongoing test program in the area of pool dynamics are properly factored into the Montague 1 and 2 design we will require that this area be resolved prior to issuance of construction permits. We advised the applicants of this and the application was revised to reflect the following actions for resolution of this area.

The applicants as part of the principal architectural and engineering criteria for the design of Montague 1 and 2 have committed to the course of action specified below for the resolution of the NRC staff's pool dynamic concerns:

(1) Small Structures Located at Elevations Less Than 19-1/2 Feet Above the Suppression Pool Surface

The applicants have made the following commitment:

These structures will either be (a) located at elevations greater than 19-1/2 feet above the pool surface or (b) designed to load profiles and associated

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time histories specified by the NRC staff (see Figures 6.2.1.6(a) and 6.2.1.6(b). The applicants may also provide the NRC staff with additional test data (which is currently available) to justify the General Electric impact load versus time profiles for small structures (e.g., piping and beams). If these profiles cannot be substantiated to the satisfaction of the NRC staff, these structures will be designed in accordance with options (a) or (b) cited above.

The NRC staff finds this commitment acceptable based on (a) it is technically feasible to locate these structures at higher elevations where pool effects are negligible and (b) sufficient test data have been made available in NEDO-11314-08 (preliminary), Information Report, Mark III Containment Dynamic Loading Conditions, to enable the NRC staff to conclude that the specified load profiles and associated time histories for small structures at their current location are acceptable.

(2) Small Structures Located at Elevations Greater Than 19-1/2 Feet Above the Suppression Pool Surface

The applicants have provided the commitment that small structures located at elevations greater than 19-1/2 feet above the pool that could be exposed to froth impingement will be designed for a load of 15 pounds per square inch and associated time history (see Figures 6.2.1.6(a) and 6.2.1.6(b).

The NRC staff finds this commitment acceptable since the specified design load and associated time history are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified criteria.

(3) Structural Protuberances from the Drywell and Containment Walls

The applicants have provided a commitment to extend these structures (e.g., the traveling incore probe station and airlocks) into the suppression pool. These structures will be designed for coincident loads due to air bubble (equal to peak drywell pressure) and pool drag (based on a pool swell surface velocity of 40 feet per second).

The NRC staff finds this commitment acceptable since the design loads are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified loads.

(4) Expansive Structures

The applicants have provided the commitment that expansive structures (e.g., the main steam line pipe tunnel and the hydraulic control unit floor) will be located at elevations greater than 19-1/2 feet above the suppression pool surface. Expansive structures located at elevations between 19-1/2 feet and

MARK III POOL SWELL IMPACT LOADS

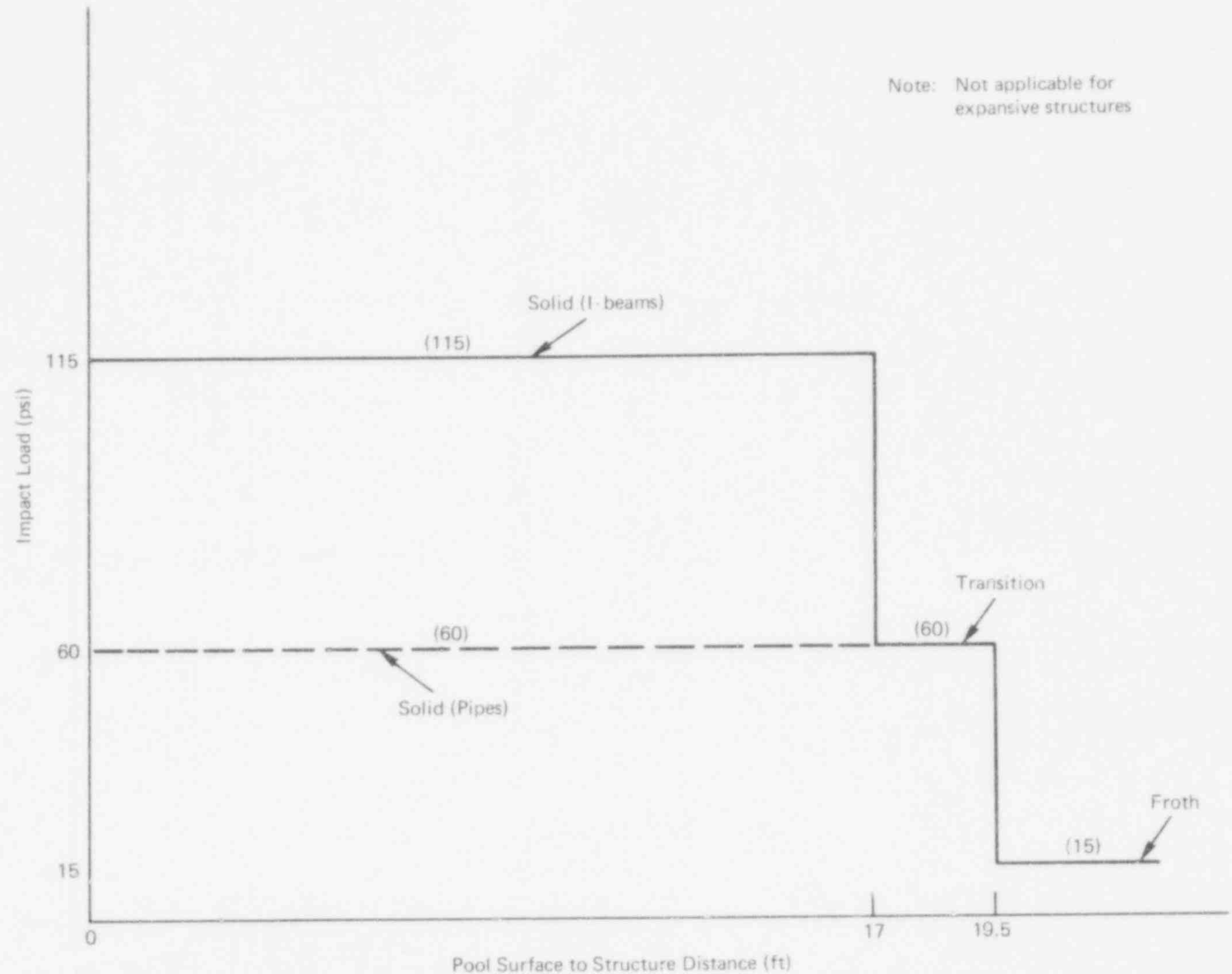

Figure 6.2.1.6(a)

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POOL SWELL IMPACT LOAD TIME PROFILES

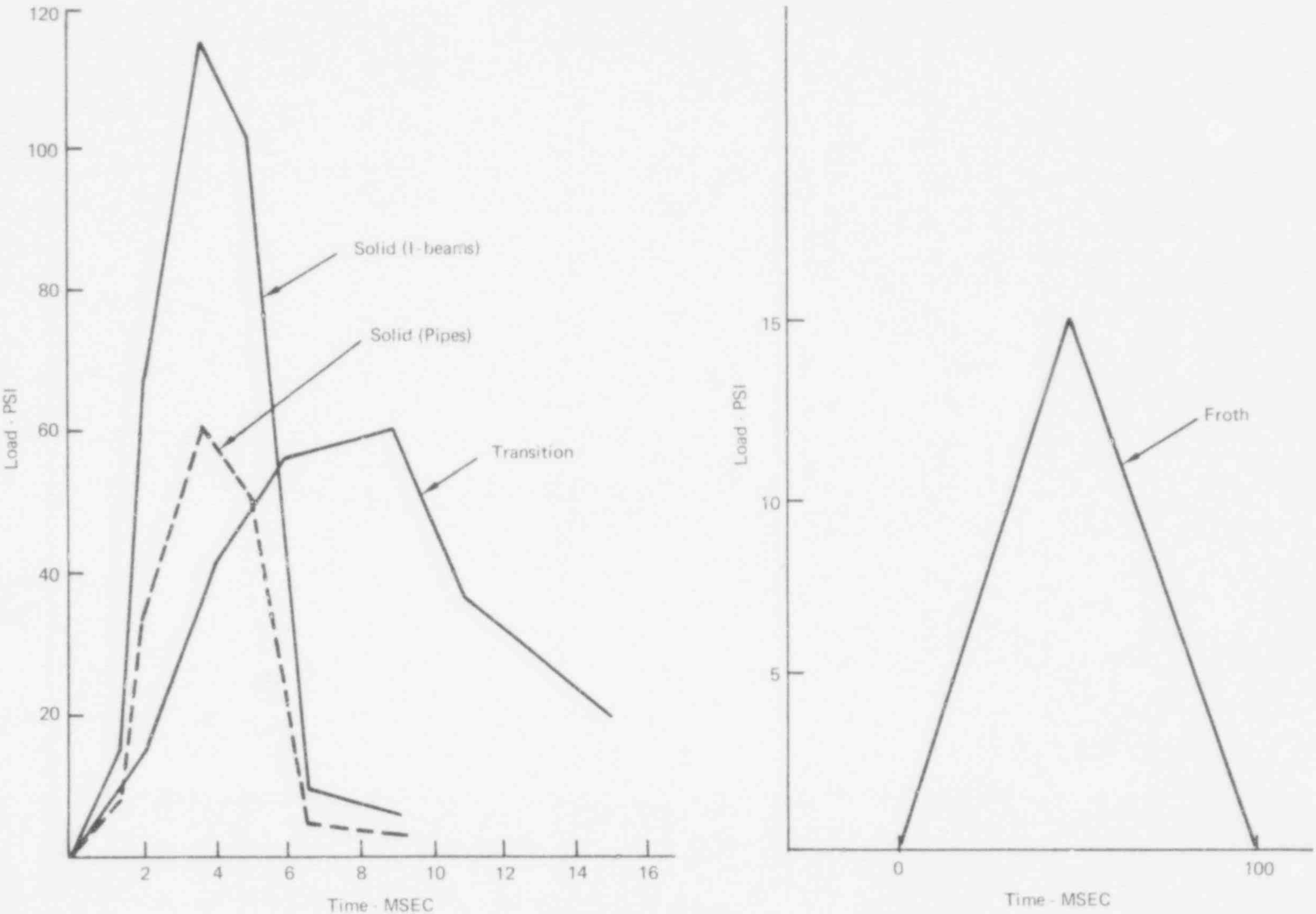

Figure 6.2.1.6(b)

30 feet above the pool will be designed for a froth impingement load of 15 pounds per square inch and associated time history (see Figures 6.2.1.6(a) and 6.2.1.6(b) and a flow pressure differential of 11 pounds per square inch. The applicants may also provide the NRC staff with additional information to justify the pool dynamic loads applied to these structures to support locating them at elevations lower than 19-1/2 feet above the suppression pool surface. However, should they be unable to justify such designs to the NRC staff, they will locate these structures at elevations greater than 19-1/2 feet above the pool surface and design these structures for the loads and associated time history cited above. At present, the Montague 1 and 2 design does not include any expansive structures located at elevations lower than 19-1/2 feet above the pool surface.

We find this commitment acceptable since the specified loads and associated time history for expansive structures at elevations greater than 19-1/2 feet above the pool are adequately supported by the test data in NEDO-11314-08 (preliminary) and it is technically feasible to design such structures to the specified loads without affecting any other aspect of the Montague design.

(5) Safety Relief Valve Loads

The applicants have provided the commitment that those structures which retain or are within the suppression pool will be designed for relief valve vent clearing loads based on the quencher design (currently under review by the NRC staff$ as this design is being used for the Montague 1 and 2. The applicants will provide the NRC staff with additional information to justify vent clearing loads for the quencher design. However, such loads must be reviewed and approved by the NRC staff prior to their use in the Montague 1 and 2 design. In the event that the vent clearing loads for the quencher design are not accepted by the NRC staff, the applicants will design these structures for the relief valve vent clearing loads based on the ramshead discharge and calculated using the analytical model as described in NEDO-20942-P. In-plant tests will be performed on Unit 1 to confirm the adequacy of these loads. If the applicants can justify to the satisfaction of the NRC staff that similar tests conducted on another Mark III design confirm the adequacy of these loads, these tests will not need to be performed on Montague 1 and 2.

We find this commitment acceptable since the analytical model appears to adequately predict relief valve clearing loads for a ramshead discharge. We also conclude that alternate designs such as quenchers are technically feasible to reduce the vent clearing loads if necessary. The in-plant testing to be performed on Montague Unit 1 is anticipated to be confirmatory in nature, and should design changes be required they could also be accommodated by a revised discharge design.

(6) Asymmetric Loads

The applicants have provided a commitment to evaluate asymmetric loads based on: (a$ the relief valve load cases listed in Section A8 of NEDO-11314-08 (preliminary); and (b) the unequal bubble load profile specified in Section 6.1.3 of NEDO-11314-08 (preliminary).

We find this commitment acceptable since the specified load cases are adequately conservative and it is technically feasible to design such structures to the specified loads.

(7) Other Structures

The applicants have provided a commitment to eliminate structures in the containment design which are not included in the preceding categories unless the design of such structures can be justified to the satisfaction of the NRC staff. In such cases, they will provide the NRC staff with additional justification to verify the bases for specification of the pool dynamic load versus time history applied to those structures. Should they be unable to demonstrate to the NRC staff that such loads are adequately conservative, these structures will be eliminated such that the design configuration of affected structures conforms to the basic Mark III design as typified by the Grand Gulf design. At present the columns supporting the main steam tunnel are the only structures in the design which fall into this category.

We find this commitment acceptable since the proposed alternatives are technically feasible as indicated by the Grand Gulf design.

(8) Other Pool Dynamic Loads

The applicants have provided a commitment that for pool dynamic loads not specifically addressed in the above criteria they will use the types, magnitudes, and combinations of loads identified in NEDO-11314-08 (preliminary) as a basis for evaluating the structural design of affected containment structures.

We find this commitment acceptable since the design loads are adequately conservative and it is technically feasible to design such structures to the specified loads.

(9) Schedule

The applicants have provided the commitment that construction of affected structures will not be initiated until at least 1979. They will defer initiation of construction of those structures for which review of additional information by the NRC staff is requested until such information has been reviewed and approved by the NRC staff. This schedule will permit completion of construction of Unit 1 at or before Fall 1985.

We conclude that the applicants' commitment provides reasonable assurance that the staff review of any additional information that the applicants may provide can be accomplished without impacting the applicants' latest date for completion of construction of the Montague Nuclear Power Station. However, the applicants do not require construction permits before 1979. All concerns involved should be completely resolved well before then. We will require such resolution prior to issuance of construction permits and will describe the final resolution of these matters in a supplement to this report.

In summary, we have reviewed the applicants' program and have concluded that the principal architectural and engineering criteria for the design of affected components and structures have been adequately described.

Based on our review of the proposed Montague 1 and 2 design and our review of these same areas of Grand Gulf (Docket Nos. 50-416 and 50-417, currently under construction), we reaffirm our conclusion that any changes which may be required as a result of our review are technically feasible without compromising safety. Such changes could include, as appropriate, relocation, local strengthening, or protection by incorporation of structures to preclude direct impingement of flow.

In addition, the staff has reviewed the comments of the Advisory Committee on Reactor Safeguards on the Mark III containment design. The Committee's comments are contained in its report on the River Bend Station. These same comments are also contained in the Committee's reports to the Commission on other BWR-6 applications.

Specifically the Committee stated that a more basic understanding of certain phenomena such as vent clearing, vent interaction, pool stratification, and dynamic and asymmetric loads on the suppression pool and other containment structures is required. It further stated that the research and development program should be expedited so that all design related issues are fully resolved prior to completion of construction of affected portions of the plant. In response to these comments, the NRC staff has expedited its review of these phenomena and has actively pursued this matter with the applicants to ensure compliance with the Committee's recommendations.

In addition, the Committee in its River Bend Station report, as well as in those for other BWR-6 applications recommended that the independent models developed by the NRC staff and its consultants "...be used to evaluate the sensitivity of key design parameters, and to elucidate additional effects noted in the experimental programs such as oscillatory phenomena."

The NRC staff is continuing with its development of an independent model to analyze the Mark III containment and fully expects to satisfy this aspect of the Committee recommendation with respect to the Mark III containment.

We consider the remaining Mark III testing and analytical programs to be confirmatory in nature and will require that these programs be completed prior to concluding our evaluation of the first application for an operating license for a Mark III plant. We believe that the concerns expressed by the Advisory Committee on Reactor Safeguards are pertinent and merit additional evaluation. However, in our judgment, they will not affect the design bases for the Montague 1 and 2 containments. We conclude that the information which has been developed to date relating to their concerns is sufficient to demonstrate the adequacy of the present design for the construction permit stage of our review.

6.2.2 Containment Heat Removal

The containment cooling mode of the residual heat removal system is used to remove heat from the suppression pool and to limit long-term containment post-accident temperatures and pressures. The system consists of two heat exchangers and three pumps. Two heat exchangers and two of the three pumps form two independent loops, and each loop is physically separated and protected to minimize the potential for single failures causing the loss of function of the entire system. The third pump is located in a separate room and can be connected to either loop. The system is designed to seismic Category I criteria.

Operating in the containment cooling mode, the pumps take suction from the suppression pool, pass it through the heat exchangers, and direct the cooled water either back to the suppression pool or into the reactor vessel. The locations of suction and return lines in the suppression pool facilitate mixing of the return water with the total pool inventory before the return water becomes available to the suction lines. Strainers are provided on the suction line inlets.

The applicants have stated in the application that adequate net positive suction head is available at the pump inlets assuming the suppression pool is at its post-drawdown level and maximum temperature, and with no credit taken for any increase in containment pressure. These assumptions are consistent with the provisions of Regulatory Guide 1.1 - Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps, and are, therefore, acceptable. Provisions are made in the containment heat removal system to permit inservice inspection of system components and functional testing of active components.

We conclude that the containment heat removal system can be operated in such a manner as to provide adequate cooling to the containment following a loss-of-coolant accident and that it satisfies the requirements of General Design Criteria 38, 39 and 40.

The applicants have also evaluated the potential for debris to clog emergency core cooling system suction lines. Each pump draws water from the suppression pool through its own suction line and strainer assembly. The applicants have shown that the potential for emergency core cooling system or containment heat removal system degradation due to plugging of the screens is minimal for the following reasons: (1) all insulation in the drywell will be of such type that it minimizes the possibility of it breaking away from piping and being carried through the drywell vent system into the pool; (2) since the suction inlets are located about three feet above the pool bottom and since the screen surface area is large, resulting in low approach velocities, there is little potential for drawing debris, either from the pool bottom or surface, to the vicinity of the inlet lines; and (3) a 50 percent plugging of screen surface area can be tolerated without significant consequences to system performance. We find this to be acceptable.

6.2.3 Secondary Containment Functional Design

The secondary containment system includes the structures and ventilation systems used to collect and process radioactive leakage from the primary containment in the event of a loss-of-coolant accident. For the Montague 1 and 2 the secondary containment structures consist of the shield building, the auxiliary building, and the fuel building. Together these structures completely enclose the primary containment. Following an accident, leakage into the shield building and auxiliary building will be collected and filtered by the standby gas treatment system prior to its release to the environment. Leakage to the fuel building will be collected and filtered by the fuel building exhaust air system.

The standby gas treatment system consists of two 100 percent capacity fan and filter trains with flow rates of 25,000 cubic feet per minute per train. The fuel building exhaust air system consists of two 100 percent capacity fan and filter trains with a rated flow of 10,000 cubic feet per minute per train. Both systems are designed to seismic Category I criteria and redundant components are separated. When actuated, these systems will maintain the secondary containment pressure to a value below atmospheric. Thus all leakage into the secondary containment volume will be routed through filters prior to release to the atmosphere.

During normal operation the annulus leak collection system maintains the annulus at a pressure of -3.0 inches water gage. Upon the receipt of an accident signal this leak collection system will shut down and the standby gas treatment system will be activated. There is a time delay in the operation of the system (and the fuel building exhaust system) while the fans are loaded onto the emergency diesels and attain rated speed. During this time period, the pressure in the annulus will rise due to inleakage and heat transfer from the primary containment shell. In addition, drawdown of the auxiliary and fuel buildings will not have occurred. Under these conditions there is increased potential for exfiltration due to winds. In the submittal of October 14, 1975, the applicants committed to provide the results of an exfiltration analysis in the Final Safety Analysis Report if the auxiliary and fuel buildings measured leak rates are determined to exceed 100 volume percent per day. We find this acceptable for the construction permit stage of our review.

Although the primary containment is completely enclosed by the secondary containment, there are systems which penetrate both the primary and secondary containment boundaries creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems of the secondary containment. A number of these lines contain physical barriers or design provisions which can effectively eliminate leakage, such as water seals, closed seismic Category I piping systems, or vent return lines to a controlled region. The integrity of these barriers is assured on the basis of the seismic and quality classification of the system. The criteria by which potential bypass leakage paths are determined have been set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

In their letter dated April 21, 1976, the applicants provided the criteria and justification of a bypass leakage of 2.5 percent of the containment design leakage rate (0.275 percent per day). We find the criteria and justification of potential bypass leakage paths are in accordance with the above Branch Technical Position CSB 6-3 and the 2.5 percent bypass leakage limit is acceptable and conclude the secondary containment design is acceptable.

6.2.4 Containment Isolation System

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary to prevent or limit the escape of fission products from a postulated loss-of-coolant accident. The applicants have specified the design bases and design criteria, as well as the isolation valve arrangements used for isolation of primary containment penetrations.

Isolation of the containment will be accomplished by automatic isolation of all fluid systems penetrating the containment that do not serve accident consequence limiting functions. Fluid lines which must remain in service following an accident for safety reasons are provided with at least one remote manual valve. The containment isolation system has been designed to the ASME Code, Section III, Class 1 or 2, and has been classified as a seismic Category I systems.

Based on our review, we conclude that the design of the containment isolation systems is acceptable and satisfies General Design Criteria 54, 55, 56 and 57.

Instrument lines that penetrate the containment were assessed in accordance with the provisions of Regulatory Guide 1.11 - Instrument Lines Penetrating Primary Reactor Containment. There are no instrument lines which are part of the protection system that penetrate primary containment. Instrument lines that are not part of the protection system and which penetrate containment have automatic isolation valves inside and outside the containment. Instrument lines which connect to the reactor coolant pressure boundary are equipped with 1/4 inch diameter orifices inside the drywell to restrict coolant release in the event of a break in one of these lines. Based on our review, we conclude that the design of the instrument lines satisfy the provisions of Regulatory Guide 1.11 and is acceptable.

The applicants have provided in their letter dated April 21, 1976 a description of the operation of the drywell and containment ventilation system. Each of the recommendations set forth in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation," have been addressed by the applicants and justification provided for any exception. We have reviewed the design and the exceptions to the recommendations of Branch Technical Position CSB 6-4 and conclude the design is acceptable.

6.2.5 Combustible Gas Control

Following a postulated loss-of-coolant accident, hydrogen may accumulate within the containment as a result of metal-water reaction between the fuel cladding and the reactor coolant and radiolytic decomposition of the post-accident emergency cooling water. The applicants have analyzed the production and accumulation of hydrogen from the above sources using the guidelines of Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." The applicants have proposed a redundant hydrogen mixing system and redundant hydrogen recombiners to limit the hydrogen concentration within the containment to below four volume percent. A backup, controlled purge system for the containment is also provided in accordance with Branch Technical Position CSB 6-2.

The hydrogen mixing system is provided to purge the hydrogen that might be produced within the drywell to the larger containment volume, thereby diluting its concentration. The system utilizes two 6-inch inlet lines, each containing two valves in series, and two 6-inch outlet lines, each containing two valves and a 600 standard cubic feet per minute compressor in series. Operation of the mixing system would not be required for about ten hours following a postulated accident at which time it would be manually actuated by the operator. The penetration size was selected so that an open mixing system line would still be within the bypass capability of the containment. Appropriate interlocks have been incorporated in the mixing system to prevent its initiation until the rate of bypassing during its operation is within acceptable limits.

The hydrogen recombiner system is a portable system consisting of two thermal recombiner units. One of the units will be located at the Montague site and the other unit at the Millstone 3 site. The required redundancy for the system will be achieved by transporting the recombiner unit from the unaffected site. Permanent piping and connections which are physically separated in the auxiliary building will be provided for two recombiners in each of the Montague units.

The applicants have stated that prototype testing of the recombiners to demonstrate their operability will be performed. We will review the testing program results at the operating license stage of our review. We will also require that the applicants provide in the FSAR the transportation and installation procedures to demonstrate that the redundant recombiner can be provided within the most limiting time frame (approximately 15 days).

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6.2.6 Containment Leakage Testing Program

The Montague containment design includes the provisions and features necessary to satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetrations and isolation valves permit individual, periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included in the proposed program of leakage rate testing are those penetrations that have resilient seals; e.g., airlocks, equipment hatches, and fuel transfer tubes.

The proposed containment leakage testing program complies with the requirements of Appendix J, 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified throughout the service lifetime and that the leakage rates will be periodically checked during service, on a timely basis to maintain such leakages within the specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through leakage paths will not be in excess of the acceptable limits specified for the site; i.e., the doses will be within 10 CFR Part 100 limits. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of Criteria 52, 53, and 54 of the General Design Criteria.

6.3 Emergency Core Cooling System

6.3.1 System Description

The subsystems of the emergency core cooling system provide for emergency core cooling during those postulated accidents where it is assumed that mechanical failures occur in the primary coolant system piping, resulting in the loss of coolant from the vessel at rates greater than the available coolant makeup capacity using normal operating equipment. The subsystems are provided in sufficient number, and with adequate independence, diversity, reliability, and redundance that, even if any single active component of the systems fails during a loss-of-coolant accident, adequate cooling of the reactor core will be maintained.

The system consists of two high pressure systems and two low pressure systems. The former are the high pressure core spray system and the automatic depressurization system. The latter are the low pressure core spray system and the low pressure coolant injection system, which is one of the modes of operation for the residual heat removal system. The system for Montague 1 and 2 are functionally identical to that of the GESSAR-238 Nuclear Island Standard Design.

All components of the emergency core cooling system are initiated by a high drywell pressure signal or a reactor vessel low water signal, except for the automatic depressurization system. Initiation of automatic depressurization system requires coincidence of both of these and a third signal, indicating pressure at the discharge of at least one low pressure emergency core cooling system pump. The system is designed to provide adequate core cooling and to limit the peak fuel rod cladding temperature for the complete spectrum of break sizes and locations up to and including the design basis loss-of-coolant accident.

The emergency core cooling system can operate independently of the offsite electrical power by using power from the onsite diesel generator and battery systems. All evaluations have been made assuming that only onsite electrical power is available. In addition, system performance capability has been shown to be adequate assuming a failure of any single active component within the system. This single failure criterion has been applied in addition to and coincident with the assumed loss of offsite power.

The high pressure core spray system consists of a single motor-driven centrifugal pump and associated system piping, valves, controls and instrumentation. The system is designed to operate from offsite power or from a its own generator. Suction is taken from the condensate tank or the suppression pool and piped to a spray sparger over the core (via two entry points at the shroud). Nozzles spaced around the sparger will spray the water over the top of the core and into the fuel assemblies. The system is designed to function over the entire range of reactor coolant system pressures and break sizes. For small breaks, the system will maintain the required reactor water level. For intermediate breaks that do not depressurize the reactor vessel rapidly, the system will depressurize the vessel. For large breaks, rapid depressurization occurs and the system cools the core in the spray cooling mode until sufficient inventory is accumulated to terminate the transient.

The pump characteristics are selected to satisfy requirements for both high pressure, low flow rate deliveries for small breaks, and low pressure, high flow rate deliveries for large breaks. When the cooling system is activated, the initial flow rate is established by reactor system pressure. As reactor pressure decreases, the flow rate will increase until the full core spray flow rate is achieved when the differential pressure between the reactor vessel and primary containment reaches 200 pounds per square inch. The pump is designed to deliver 6110 gallons per minute at 200 pounds per square inch difference and 1465 gallons per minute at 1140 pounds per square inch difference, and has a shutoff head of 1370 pounds per square inch difference.

The automatic depressurization system is designed to reduce the reactor pressure so that flow from the low pressure coolant injection and low pressure core spray can enter the reactor to cool the core and limit the fuel cladding temperature. The system utilizes eight of the 19 safety-relief valves in the pressure relief system. Automatic opening of these valves requires coincident signals of reactor vessel low water and high drywell pressure along with a high discharge pressure indication on any low pressure coolant injection or low pressure core spray pump, but only after a timer delays operation of the automatic depressurization system relief valves for two minutes. If the operator determines that the initiation signal is false or depressurization is not required, the timer may be recycled. The automatic depressurization system is redundant to the high pressure core spray and is only required if the high pressure core spray cannot maintain reactor water level following an accident. As with the high pressure core spray, the depressurization function of the automatic depressurization system is not required for large breaks.

The low pressure core spray system consists of a motor-driven centrifugal pump (that can be powered by either normal offsite power or the standby ac power system); a spray sparger in the reactor vessel; and piping, valves, instrumentation and controls to convey water from the suppression pool to the sparger.

The high pressure core spray system operating in the low pressure mode serves as a redundant core spray loop to the low pressure core spray loop. The low pressure core spray system protects the core in the event of a large break in the reactor coolant pressure boundary and when the high pressure core spray is unable to maintain the required reactor vessel water level. Such protection extends to the small break in which the automatic depressurization system or high pressure core spray has operated to lower the reactor vessel pressure to the operating range of the high pressure core spray. The low pressure core spray pump is designed to deliver 6110 gallons per minute at 122 pounds per square inch difference and has a shutoff head of 289 pounds per square inch difference.

Since the number of fuel assemblies and the diameter of the core has changed relative to previous designs, spray distribution tests will be performed on a simulation of the GESSAR reactor to assure that an adequate amount of spray reaches every assembly. These tests will also be applicable to the Montague reactors. General Electric states that no significant differences are expected from other core geometries previously tested for spray distribution. General Electric is to provide the results of these tests in a topical report when they are completed. We will review that report prior to the operating license stage of review. This approach is acceptable for the construction permit stage of our review for Montague 1 and 2.

The low pressure coolant injection system consists of three motor-driven centrifugal pumps (that can be powered by either normal offsite power or the standby, onsite alternating-current power system), associated piping, valves, controls and instrumentation. Each pump injects water from the suppression pool through a nozzle in the core shroud into the space between channel boxes over the active core. The suppression pool suction, vessel injection nozzle and connecting piping for each pump are separate and independent. Two of the pumps also function as residual heat removal system pumps. These two pumps receive power from different alternating current power buses. One of these buses also supplies power to the third low pressure coolant injection pump, and the second bus supplies power to the low pressure core spray pump.

The low pressure core spray system provides cooling water following all loss of coolant accidents except those resulting from small breaks that can be controlled by the high pressure core spray system. The low pressure coolant injection system is redundant to the low pressure core spray system. Each low pressure coolant injection pump delivers 7100 gallons per minute at 26 pounds per square inch difference and has a shutoff head of 225 pounds per square inch difference.

As in the previous plant designs, the Montague design has the capability to use the low pressure coolent injection pumps to spray water into the containment. Diversion of these pumps after a loss-of-coolant accident is automatic when required. In previous designs, an interlock prevented diversion of the pumps if the vessel water level was below 2/3 the active core height. In the proposed arrangement for this plant, this interlock will not be provided but instead an interlock preventing pump diversion to containment spray until 10 minutes after an accident will be provided. General Electric has presented an analysis of the performance of the emergency core cooling system over the complete spectrum of breaks assuming that two low pressure coolant injection pumps are diverted from core cooling to containment spray 10 minutes after an accident occurs. We have reviewed the information submitted and have determined that the performance of the emergency core cooling system is not significantly affected by the transfer of two pumps from core cooling to containment spray. We conclude that the proposed transfer after 10 minutes is acceptable.

We raised a concern regarding the overall role of manual actions required to mitigate the consequences of a loss-of-coolant accident of GESSAR-238 Nuclear Island Standard Design. General Electric has agreed to provide the necessary information for our review prior to the time that final designs are available. We consider this commitment by General Electric to be acceptable for the construction permit stage of our review for Montague 1 and 2 because of the applicants' commitment to adopt the GESSAR resolutions.

In our letter of May 1974 to General Electric, we identified certain outstanding issues concerning or related to the emergency core cooling system. These issues will be resolved on the Montague docket in our review of the Montague Final Safety Analysis Report. The subjects covered in our letter of May 1974 were: (1) a list and description of the purpose of pre-operational and startup tests of certain (mostly emergency core cooling) systems; (2) justification of applicability of referenced reports to BWR-6; (3) description of methods used and results of blowdown load calculations on reactor internals; (4) a list of all emergency core cooling system related valves operated by containment isolation signals; (5) details of the calculational methods used to show that net positive suction head requirements of emergency core cooling system are met; and (6) quantitative details of the main steam line radiation detector's ability to detect failed fuel. The resolution of these items will be discussed during our review of the Montague Final Safety Analysis Report. Resolution of these items is not required at this construction permit stage of our review. Accordingly there is reasonable assurance that any related safety questions will be resolved at the operating license stage of review.

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6.3.2 Performance Evaluation

In Section 6.3.3 of the application, the applicants provide an analysis applicable to Montague 1 and 2 which complies with the requirements of Section 50.46 and Appendix K to 10 CFR Part 50. The analysis was performed using evaluation models as described in NEDE-20566 (draft) submitted in August 1974, and the Refill/Reflood Calculation (supplement to the SAFE code description) transmitted to the staff by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974. General Electric has submitted an additional report by letter from G. L. Gyorey to V. Stello, Jr. dated August 25, 1975, that discusses the way in which the REFLOOD model is used in the analysis of boiling water reactors with in-shroud low pressure coolant injection. The background of the staff review of the emergency core cooling system models is described in the staff Safety Evaluation Report issued in connection with Order dated December 27, 1974 for operating jet pump boiling water reactors. The bases for acceptance of the principal portions of the General Electric Evaluation Model are set forth in the staff's Status Report of October 1974 and the supplement to the Status Report of November 1974 which are referenced in the Safety Evaluation Report, dated December 27, 1974. Together, the December 27, 1974 Safety Evaluation Report on operating plants, the Status Report and its supplement describe the basis for the staff's acceptance of the evaluation model. The General Electric evaluation model in combination with the plant specific parameters constitutes an acceptable evaluation in conformance with Appendix K to 10 CFR Part 50 and is applicable to the Montague Nuclear Power Station.

During the course of our review, we concluded that additional break sizes should be analyzed to substantiate the break spectrum curve. We also requested that other break locations (i.e., steam line, feedwater line, and core spray) be studied to substantiate that the limiting break location was the recirculation line. As part of the loss-of-coolant accident analysis, additional BWR-6 single failure sensitivity analyses were performed to evaluate the effects of a single failure that could cause any manually controlled electrically operated valve to move to a position that adversely affects the emergency core cooling systems. The analyses showed that these failures are less severe than those considered for the emergency core cooling system analysis.

We also investigated the effects of flooding of the containment caused by a postulated loss-of-coolant accident. By letter, dated August 11, 1975, General Electric submitted the results of a study on emergency core cooling system valves within the containment. The results show that all valve motors which must be operable during and after a loss-of-coolant accident are located outside the containment and will not become submerged due to the occurrence of an accident. Therefore, neither the short-term requirement nor the long-term cooling capability is affected by submergence effects. The applicants have referenced the GESSAR-238 Nuclear Island Standard Design as the lead plant design concerning the emergency core cooling system and will incorporate all appropriate changes made to GESSAR-238 Nuclear Island Standard Design by amendments to the Montague 1 and 2 docket.

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The results of the Appendix K calculation for Montague 1 and 2 show a peak cladding temperature of 2180 degrees Fahrenheit; a peak local oxidation of two percent, and a maximum core average metal water reaction of 0.25 percent for the worst large break assuming a failure of the low pressure coolant injection diesel. A peak cladding temperature of 1680 degrees Fahrenheit at a break area of 0.3 square feet assuming the failure of a low pressure core spray diesel; and a peak cladding temperature of approximately 1520 degrees Fahrenheit at 0.1 square feet assuming a failure of the high pressure core spray diesel, were calculated for the intermediate and small breaks (based on flat local peaking).

We have reviewed the evaluation of emergency core cooling system performance submitted by the applicants for Montague, 1 and 2 and conclude that the evaluation was performed wholly in conformance with the requirements of 10 CFR 50.46(a). The system performance assures conformance with: (1) the peak cladding temperature limit of 2200 degrees Fahrenheit; (2) the maximum cladding oxidation limit of 17 percent of total cladding thickness before oxidation; (3) the maximum hydrogen generation core wide limit of one percent of the total metal in the cladding thickness before oxidation; (4) the core geometry remaining amenable to cooling; and (5) the long-term cooling requirement of maintaining acceptable core temperatures and decay heat removal.

An evaluation was not provided for emergency core cooling system performance during reactor operation with one recirculation loop out of service. Therefore, reactor operation under such conditions will not be authorized until the necessary analyses have been performed, evaluated, and determined to be acceptable by the staff.

With regard to our concern relating to recirculation valve closure during a loss-of-coolant accident, we have reviewed this on a generic basis on the GESSAR-238 Nuclear Island docket (STN 50-447). The results of the sensitivity study which considered this event were submitted to the staff by letter from A. J. Levine to V. Stello dated April 25, 1975. The results show that the consequences of the single failure are less severe than the other single failures considered and are, therefore, acceptable.

In summary, we conclude that Montague 1 and 2 meets all of the criteria of Appendix K to 10 CFR 50 and is acceptable.

The above evaluation of Montague 1 and 2 was made for that design which does not include the prompt relief trip. If our review of the prompt relief trip, which is ongoing, finds the system function to be required, then the evaluation will also have to include review of the prompt relief trip effects on the emergency core cooling system performance prior to a decision for issuance of construction permits. Further discussion of this matter is provided in Section 15.2.

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6.4 Habitability Systems

The emergency protective provisions of the control room related to the accidental release of radioactivity or toxic gases are evaluated in this section. Relevant portions of the control room ventilation system are described briefly here, and are described and evaluated more fully in Section 9.4.

6.4.1 Radiation Protection Provisions

The applicants propose to meet General Design Criterion 19 by use of concrete shielding and by installing two remote fresh air inlets to provide an assured source of clean air for pressurization. In addition, the design incorporates a redundant 2,000 cubic feet per minute charcoal filter train that processes the make up air to further ensure a habitable environment within the control room zone.

The system will consist of a split inlet arrangement where the make up air is taken from one of three sources, the normal inlets, located on each of the control buildings, an inlet 1000 feet northeast of Unit 1 containment, and an inlet 1000 feet south of Unit 2 containment. Each of the remote inlets will be approximately 175 feet inside the nearest security fence. Radiation detectors will monitor each inlet. Upon receipt of a high radiation signal from a normal air intake, an alarm will be sounded in the control room. The normal inlets will be automatically isolated while the make up air supply will be taken from the two remote inlets and will be automatically diverted through the charcoal filter. The remote inlet which exceeds a preset contamination level will be prevented from opening by a permissive type interlock.

We have estimated the doses to the control room operator after a design basis loss-of-coolant accident. In our analysis, we assumed that the operating inlet was periodically exposed to contamination due to delay in manual operation as follows:

Time Period After Accident	Period of Exposure (Hours)
0 - 8 hours	2.0
8 - 24 hours	0.5
1 - 4 days	0.5 per day
4 - 30 days	0.5 per day

The object of this analysis was to evaluate manual operation of the inlets versus automatic operation. Our preliminary analysis indicates that manual selection may result in excessive contamination of the control room. For this reason, and to also eliminate the necessity of diverting the operator's attention in an emergency situation, the applicants have modified the system so that inlet selection would be automatic based on radiation indications at the remote inlets. The present damper configuration ofthe remote intakes does not meet the single active failure criterion in that if a damper fails open, isolation of the contaminated inlet cannot be

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accomplished. However, the location of these dampers will permit an operator to manually override a failed damper without leaving the pressurized area. We find this acceptable.

We conclude that the radiation protection system for the control room area is acceptable and meets the guidelines of Criterion 19 of the General Design Criteria.

6.4.2 Toxic Gas Protection Provisions

The applicants have indicated that no chemicals will be stored on or in the vicinity of the site that would pose a potentially hazardous condition inside the control room if accidentally released. Chlorine will not be stored on the site. We conclude that special protective provisions against toxic gases are not required.

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7.0 INSTRUMENTATION AND CONTROL

7.1 General

The proposed designs for the instrumentation and control systems were reviewed utilizing: (1) the Commission's General Design Criteria (July 1971); (2) the various Institute of Electrical and Electronics Engineers Standards including the "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE Std 279-1971); and (3) the applicable Regulatory Guides for Power Reactors, as the bases for evaluating their adequacy. Specific documents relied on in our review are listed in Appendix B.

The applicants have chosen to utilize the instrumentation and control design for the General Electric boiling water reactor as described in the General Electric Standard Safety Analysis Report for the GESSAR-238 Nuclear Island Standard Design and have incorporated the applicable GESSAR docketed information into the Montague Preliminary Safety Analysis Report via blue pages. In addition, the applicants have chosen to adopt the resolutions achieved or to be achieved by General Electric and the NRC staff on the GESSAR-238 docket (STN 50-447). Therefore, the review of the protection and control systems was accomplished by comparing the design proposed for the Montague units with the design proposed for the GESSAR-238 design for which a Preliminary Design Approval was issued in December 1975. This comparison, combined with the applicants' commitment to adopt the resolutions achieved on the GESSAR-238 docket, allowed us to proceed with a review which concentrated on those aspects of the proposed design which are unique to the Montague facility. The specific areas of continuing review effort on the GESSAR-238 docket are outlined in the following sections of this report and in Section 1.8. We expect all outstanding matters on GESSAR-238 to be resolved near the end of this year. Further, since construction permits for Montague 1 and 2 are not needed before 1979, we intend to issue a supplement to this report that resolves all outstanding matters near that time.

7.2 Reactor Trip System

The design of the Montague reactor trip system is to be identical to the design of the GESSAR reactor trip system. The GESSAR reactor trip system is a new design proposed by the General Electric Company. The NRC staff is conducting a review of the proposed preliminary design as part of the post-Preliminary Design Approval for the GESSAR-238 docket. The conceptual design for the reactor trip system consists of four identical divisional logic channels with each of these four channels receiving input signals from four sensors per monitored variable. Each of the four sensors associated with each monitored variable provides an input signal to each of the four divisional logic channels through isolation devices. The divisional logic channels utilize "2-out-of-4" coincidence logic for each set of four signals to generate a trip signal, i.e., when 2-out-of-4 signals for a given input variable exceed the trip set point, a divisional logic output signal is produced. The divisional logic output signals are the input signals for the actuator logics which control the electric power for the scram pilot solenoid valves. The actuator logics utilize "1-out-of-2 taken twice" logic to initiate a reactor trip by de-energizing the scram pilot solenoid valves. The conceptual design arrangement described above is illustrated in Figures 7.2-3a through 7.2-3f of the Montague application.

The manual scram logic and back-up scram valve logic will be "1-out-of-2 taken twice" as used in previous boiling water reactor plant designs. The functional arrangement of the solenoid-operated pilot scram valves, the solenoid-operated back-up scram valve, and the air-operated scram valves will also remain the same as in previous plant designs.

The preliminary design of the reactor trip system for Montague 1 and 2 and GESSAR-238 is presently under review and all outstanding items associated with the reactor trip system are expected to be resolved during this review which is expected to be completed by the end of this year.

Based on the applicants' commitment to adopt the final resolutions developed on GESSAR-238, we conclude that an acceptable design of the reactor trip system will be available prior to the need for construction permits. We will report on the resolution of these resolved matters in a supplement to this report.

7.2.1 Anticipated Transients Without Scram

The NRC staff's requirements with respect to anticipated transients without scram are provided in the staff's technical reports, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270 dated 1973 and "Status Report on Anticipated Transients Without Scram for General Electric Reactors" dated December 9, 1975. The applicants have been sent a letter requesting that the staff's positions provided in the above December 9, 1975 status report be addressed for the Montague 1 and 2 and submitted for our review by June 30, 1977. We are continuing our review of this matter and will require that any changes that are indicated to be needed be incorporated into the Montague 1 and 2 designs.

7.2.2 Safety Interfaces with the Reactor Protection System

In response to our request regarding certain reactor protection system trip signals that are derived from the pressure regulator and turbine control system, the applicants responded that the steam bypass valve position switches have been deleted (from the reactor protection system) by General Electric. We are pursuing the significance of all turbine related inputs to the system on the GESSAR-238 docket. The applicants have committed to adopt the resolution achieved on GESSAR-238 with

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regard to the use and qualification of the turbine related reactor trips. The staff finds this commitment acceptable since it will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3 Engineered Safety Feature Systems

The design of the instrumentation and control systems for the engineered safety features actuation systems is to be identical to the instrumentation and control design for the GESSAR-238 systems. The design proposed by General Electric for GESSAR is functionally the same as that of previous plant designs except for a few design changes. However, the instrumentation and control hardware (i.e., the actuators, logic and sensors) is not similar to any previous plant design. Therefore, the staff has determined that a review of the proposed preliminary design should be a part of the post-Preliminary Design Approval review being conducted for the GESSAR-238 docket.

As stated in the GESSAR-238 Safety Evaluation Report, the proposed conceptual design was reviewed and found acceptable. The preliminary design is presently under review. Areas of concern are to be resolved during this detailed review. We find the applicants' commitment to adopt the resolution developed on the GESSAR-238 docket acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3.1 Emergency Core Cooling Systems

7.3.1.1 High Pressure Core Spray System

The instrumentation and control system for the high pressure core spray system for Montague 1 and 2 is identical to the GESSAR-238 system. The preliminary design is presently under review. However, the major areas of concern between the staff and General Electric have been resolved.

Based on the applicants' commitment to adopt the GESSAR-238 resolution, we conclude that the proposed instrumentation and control system for the system is acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permit for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3.1.2 Automatic Depressurization System

The instrumentation and control system for the automatic depressurization system for Montague 1 and 2 is identical to the GESSAR-238 design. The major areas of staff concern are generic issues and resolution during our review of the preliminary design on the GESSAR-238 docket is required by the staff. This review is presently under way.

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Based on the appl nts' commitment to adopt the resolution achieved on the GESSAR-238 docket we conclude that the proposed instrumentation and control system for automatic depressurization system is acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3.1.3 Low Pressure Core Spray and Low Pressure Coolant Injection Systems

The instrumentation and control system for the low pressure core spray system and the low pressure coolant injection mode of operation of the residual heat removal system is identical to that of the GESSAR-238 design. The major areas of concern are being resolved during the detailed review of the preliminary designs for these systems.

Based on the applicants' commitment to adopt the GESSAR-238 resolution, we conclude that the proposed instrumentation and control system for the low pressure coolant injection and low pressure core spray is acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3.2 Containment and Reactor Vessel Isolation Control System

The instrumentation and control system for the containment and reactor vessel isolation control system is identical to that of the GESSAR-238 design. The main steam line isolation portion of this system is also identical to the GESSAR-238 design and the specific control arrangement for these valves is being reviewed in the course of our review of topical report APED-5750, "Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves."

Based on the applicants' commitment to adopt the GESSAR-238 resolution and to incorporate the resolution developed during our review of the above cited topical report, we conclude that the proposed instrumentation and control system for this system is acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.3.3 Combustible Gas Control

The original system proposed for the control of combustible gas (hydrogen) inside the containment was unacceptable. The system relied on operator action within a relatively short period of time after an accident and the staff required such action to be automatic. The latest proposed system is designed with allowance for new hydrogen concentration versus time predictions. Based on a recently revised guidance (Regulatory Guide 1.7-Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident), the amount of time allowed until actuation is required has been extended to approximately 10 hours. Therefore, manual initiation is acceptable.

Based on the above criterion and the fact that the system is similar to other boiling water reactor designs, the staff concludes that the system is acceptable.

7.3.4 Standby Gas Treatment System

The standby gas treatment system is an engineered safety feature which serves to limit the release of particulate and gaseous radioisotopes within the guidelines of 10 CFR Part 100. The proposed system will have full redundancy and will be automatically initiated by the same signals which initiate the core cooling function (i.e., low reactor water level and high drywell pressure), and in addition, by high radiation signals in various plant areas.

The applicants have stated that the standby gas treatment system will function in a manner similar to the system for the River Bend Station. Based on the information presented and the comparison with River Bend, the staff finds the proposed system acceptable.

7.3.5 Auxiliary Support Systems

The auxiliary support systems to the engineered safety feature systems for Montague 1 and 2 consist of:

(1) Standby service water system (including portions of the reactor plant component cooling water system and the chilled water system.

(2) Ultimate heat sink system.

(3) Control building atmosphere control system.

(4) Containment ventilation system.

(5) Standby generator support system.

(6) Emergency core cooling systems fill system.

The instrumentation and control systems for these systems were reviewed to determine that each has sufficient redundancy and independence to provide the required support to the engineered safety features system. Each system was compared with similar systems in other plants including those which are currently under review or those for which construction permits have been issued.

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The staff concludes that the design of the instrumentation and control system for the auxiliary support systems conform to all applicable regulations, guides, branch technical positions, and industry standards and are acceptable.

7.3.6 Regulatory Guide 1.53 - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Sysems

With reference to IEEE Std 379 "IEEE Trial Use Guide For the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems", the applicants have stated their interpretation of the statement ".....any and all combinations of non-detectable failures" in IEEE Std 379-1972 Paragraph 3(3). In the applicants' explanation, there is a staff concern over the possibility of not detecting certain failures. Specifically, the applicants state that "operational tests of final actuation using parallel wiring would not detect a single failure of one leg of the parallel circuit since the unfailed legs of the parallel circuit would maintain circuit function." The staff believes that interpretation can be used to develop acceptable designs. We will review the electrical wiring details at the operating license stage of our review to assure that the safety criteria are not violated.

7.3.7 Testing of Engineered Safety Feature Systems and Auxiliary Support Systems

In the original application submittal, the applicants proposed to test the automatic responses of all engineered safety features systems periodically with the exception of the standby service water system. The applicants stated that the testing of this system would adversely affect the plant. In response to a staff request for justification of this exception, the applicants revised the design to allow periodic (on-line) testing of all engineered safety features systems including the standby service water system. Based on these changes, we find the proposed test provisions acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 Reactor Core Isolation Cooling System

The instrumentation and control system for the reactor core isolation cooling system is identical to the GESSAR-238 system. In response to a staff position, the applicants have identified this system as an engineered safety feature system as did General Electric for the GESSAR-238 docket.

Based on the applicants' commitment to adopt the GESSAR-238 design, we have concluded that the instrumentation and control for this system is acceptable.

7.4.2 Standby Liquid Control System

The instrumentation and control system for the standby liquid control system is identified as being similar to the Zimmer design. This system is also the same as the proposed GESSAR-238 design.

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The applicants have chosen to utilize the design provided on the GESSAR-238 docket and therefore, we find the system acceptable.

7.4.3 Safe Shutdown From Outside the Control Room

General Design Criterion 19 requires that nuclear power plants have the ability to achieve reactor shutdown coincident with an evacuation of the main control room. We have reviewed the applicants' proposed methods for implementing these requirements including the provisions to prevent a possible compromise of divisional circuit separation and unauthorized access to the panel.

We find the applicants' proposed methods to be in accordance with the requirements and therefore acceptable.

7.5 Safety-Related Display Instrumentation and Indication of Bypass

We have been reviewing on a generic basis the requirements for display instrumentation to diagnose the plant's status during the course of an accident. It is anticipated that these requirements will be further identified and defined in a forthcoming regulatory guide on the subject. Pending issuance of the regulatory guide, we require that the post-accident instrumentation for Montague 1 and 2 be qualified for the appropriate accident environment. We will require that all safety-related instrumentation be: (1) redundant with at least one channel recorded; (2) energized from onsite power supplies and; (3) in compliance with the applicable requirements of IEEE Std 279-1971.

The applicants have documented their intent to conform to these requirements and therefore, we find these design criteria acceptable for the construction permit stage of our review.

Our review of the provision for indication of bypassed or inoperable status conditions of plant safety systems utilized the recommendations of Regulatory Guide 1.47 - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems. The applicants have stated their intention to conform to the Regulatory Guide, although a preliminary design has not been provided. We conclude that the commitment is acceptable and there is reasonable assurance that the applicants can develop an indication system that conforms to Regulatory Guide 1.47.

7.6 Other Instrumentation Systems Required for Safety

The applicants have identified the following instrumentation systems as being identical to those in GESSAR-238. GESSAR-238 in turn identifies these systems as being similar to those in recent boiling water reactor plants that have been authorized for construction.

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(1) Refueling interlock.

(2) Reactor vessel instrumentation.

(3) Process radiation.

(4) Area radiation.

(5) Reactor water clean-up.

(6) Leak detection system (portion).

(7) Nuclear steam supply system computer system.

(8) Neutron monitoring.

We are presently reviewing these systems on the GESSAR-238 docket as post-Preliminary Design Appoval matters. The staff is making an effort to review these systems with a minimum amount of reliance on previous boiling water reactor plant design reviews. The results of our post-Preliminary Design Approval review of the above systems will be included in a supplement to the GESSAR-238 Safety Evaluation Report.

The applicants' commitment to accept the generic resolutions on GESSAR-238 in these areas is acceptable. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.6.1 Rod Control and Information

The proposed BWR-6 rod control and information includes a new rod pattern control system and a rod position indication system. The staff is in the process of reviewing these systems in detail on GESSAR-238 because of the safety significance given them in limiting the consequences of a rod drop accident.

The applicants have included the systems in the group of instrumentation and control systems which are to be identical to the GESSAR-238 systems. The staff finds this approach acceptable for the construction permit stage of our review. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

7.7 Control Systems Not Required for Safety

The applicants have stated that the feedwater control system, the recirculation flow control system, and the pressure regulation control system are similar to

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those in previously acceptable boiling water reactor plants, and in addition, are to be identical to the GESSAR-238 systems. We conclude, therefore, that these designs are acceptable.

7.8 Seismic Qualification

The applicants state in Section 3.10 of the application that seismic Category I instrumentation and electrical equipment will be designed to withstand the effects of the safe shutdown earthquake without functional impairment. All Class IE equipment will be qualified in accordance with the requirements of IEEE Std 344-1971, and Branch Technical Position Electrical Instrumentation and Control Systems Branch-10 in Appendix 7A of the Standard Review Plan. We conclude that these commitments provide assurance that the proposed equipment important to safety, will be seismically qualified in an acceptable manner.

7.9 Environmental Qualification

The applicants have stated that Class IE equipment will be qualified in accordance with IEEE Std 323-1974, but that exceptions to the aging requirements for certain equipment may be necessary due to state-of-the-art problems such as lack of data or analytical techniques and inability to obtain competitive bids. The applicants propose that in these cases, prior qualification can be utilized and that one of the following methods singularly or in combination will validate the qualification of that equipment:

(1) Analyses based upon environmental tests.

(2) Operating experience (taking into consideration inservice inspection,) periodic tests, and preventive maintenance.

(3) Type tests using qualitative aging techniques (e.g., environmental cycling and operational cycling elevated stress techniques).

(4) On going or pacing tests.

The applicants' commitment to IEEE 323-1974 and to provide to the staff prior to procurement of Class IE equipment justification for any specific exceptions on aging, is acceptable.

7.10 Turbine Overspeed Protection

We have reviewed the applicants' proposed turbine overspeed protection system. The overspeed protection system is comprised of redundant mechanical hydraulic and electro-hydraulic channels. Physical separation between redundant channels is provided and there are provisions for periodically testing the system while the plant is at power.

We have concluded that the provisions for turbine overspeed protection are acceptable.

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8.0 ELECTRIC POWER SYSTEMS

8.1 Introduction

The Commission's General Design Criteria 17 and 18, Regulatory Guides 1.6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems and 1.9 - Selection of Diesel Generator Set Capacity for Standby Power Supplies, and IEEE Std 308-1971 were utilized as the primary bases for evaluating the adequacy of the electric power systems for the Montague 1 and 2. Specific documents used in the review are listed in Appendix B.

8.2 Offsite Power System

The main switchyard for Montague 1 and 2 consists of a 345 kilovolt ring bus that is to be converted to a breaker-and-a-half arrangement upon completion of Unit 2. The switchyard will provide terminal facilities for the outputs of the main generators of both units, four offsite transmission lines, and two lines each of which will serve the two system station service transformers of each of the two units. The four offsite transmission lines to the switchyard are to be routed to minimize the likelihood of a simultaneous failure.

The system station service transformers (two per unit) provide the preferred offsite power source for the two units. In addition, the main generator of each unit is connected to two unit station service transformers to provide power during normal operation.

The station service bus system for each unit will consist of four 13.8 kilovolt buses and two normal 4.16 kilovolt buses which serve the non-safety auxiliary loads, and three standby 4.16 kilovolt buses which serve the safety related loads. The four 13.8 kilovolt buses are fed from either the unit station service transformers or the system station service transformers. Thus each 13.8 kilovolt bus is fed from the normal and preferred offsite source. Automatic switching to the preferred source is provided on failure of the normal source. The two normal 4.16 kilovolt buses are fed similarly but also are provided with a circuit from the shutdown transformer. This circuit is physically and electrically independent from the other circuits to meet the requirements of General Design Criterion 17.

The original submittal for Montague 1 and 2 contained provisions for this alternate offsite source to be from a single shutdown transformer shared between the two units. In response to a staff request, which pointed out the requirement that loss of this transformer would require shutdown of both units within a short period of time, the applicants elected to provide a separate shutdown transformer for each unit and

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thus eliminate any possible concerns about sharing. This arrangement provides each normal 4.16 kilovolt bus with a feed from the normal source, preferred source and alternate offsite source.

The three standby 4.16 kilovolt buses are fed from the two normal 4.16 kilovolt buses, with one normal 4.16 kilovlt bus feeding one standby 4.16 kilovolt bus and the other normal 4.16 kilovolt bus feeding two 4.16 kilovolt standby buses, one of which is the high pressure core spray system bus.

We have reviewed the applicants' plans for testing the offsite power system and find that the requirements of General Design Criterion 18 are satisfied.

We conclude that the design of the offsite power system source meets the requirements of the applicable regulations, guides, technical positions and industry standards and is acceptable.

8.3 Onsite Power Systems

8.3.1 Alternating-Current Standby Power Source

In the event that all sources of normal and offsite power are lost for a unit, the auxiliaries essential to safe shutdown are supplied by standby diesel generators. There are three standby generators for each unit. Each standby generator is connected to one of the three 4.16 kilovolt standby buses to which the safety related systems are connected. The loads are grouped and identified by the applicants as Division I, II and III loads. Each standby generator set is operated independently of the others and is, except for testing, disconnected from the utility power system. There is no sharing of standby generator between the two nuclear units.

Each standby generator is physically independent, of the other generators. Each generator is located in a seismic Category I structure which is designed to withstand earthquakes and to protect the standby generators against the effects of tornadoes, floods, hurricanes and tornado generated missiles.

The high pressure core spray generator set (Division III) is presently under generic review by the NRC staff in conjunction with General Electric Topical Report NEDO-10905, "High-Pressure Core Spray System Power Supply Unit." The staff has concluded that NEDO-10905 is presently not acceptable for qualifying the high pressure core spray standby generator. General Electric is in the process of satisfying the staff's concerns regarding this topical report. The applicants when notified of the status of the topical report review agreed to adopt that generic resolution of the outstanding concerns that is developed in the course of our review of the above cited topical report. This commitment will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

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The applicants have stated, in response to a request by the staff, that should the Division I and II standby generators be of a type not previously qualified for standby power service at a nuclear generating station, they will be qualified in accordance with the following which meets the staff's position in this regard.

(1) At least two full-load and margin tests will be performed on each standby generator set to demonstrate the start and load capability of the units with some margin in excess of the design requirements. Proposed full-load and margin testing will be evaluated on an individual case basis to take account of the differences in unit design.

(2) Prior to initial fuel loading, at least 300 valid start and load tests will be performed with no more than three failures allowed. At least 90 percent of these start tests will be made from design cold ambient conditions (design hot standby conditions if standby temperature control system is provided) and 10 percent from design hot equilibrium temperature conditions. This would include all valid tests performed offsite. A valid start and load test is defined as a start from the specified temperature conditions with loading to at least 50 percent of continuous rating within the required time intervals, and continued operation until temperature equilibrium is attained.

(3) A failure rate in excess of one per hundred tests will require further testing as well as review of the system design adequacy.

The standby generators will be used for supplying power only to the loads on the standby power system except as necessary to load the standby generators for periodic testing.

The fuel oil storage and transfer facility for each diesel generator will be designed to provide sufficient fuel oil for at least seven days of continuous operation of the diesel carrying the full emergency load of the associated generator.

Based on our review, we conclude that the design of the standby alternating-current power system meets General Design Criteria 17 and 18 and Regulatory Guides 1.6 and 1.9 and is therefore acceptable.

8.3.2 Direct Current (D-C) Power System

There are a total of five ungrounded, 125 volts safety related direct-current battery systems; two systems for Division I, two systems for Division II, and one system for Division III. There is also a direct-current ungrounded non-safety related system.

Each system includes a battery charger, a lead acid battery, a battery distribution switchboard, a subordinate battery distribution panel, local and control room instrumentation, and alarm facilities. Safety related chargers are powered from the standby power system of their own division.

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The chargers, distribution switchgear, and certain subordinate equipment such as uninterruptible power supplies, will be located in separate ventilated, temperature controlled rooms shared with other items of electrical equipment. Each of the batteries will be located in a separate ventilated temperature controlled room. The enclosures for safety related equipment are designed to seismic Category I criteria. The other electrical equipment in the room will be safety related equipment and of the same division.

Battery room temperatures will be maintained between 60 and 80 degrees Fahrenheit. Therefore, temperature compensations of battery capacity is not required.

Each battery has the capacity to supply all its connected safety related direct and alternating-current vital loads for a minimum of two hours upon interruption of alternating-current power supply to the battery charger.

Additional independent direct-current service power is provided for the unit switchyard.

Based on our review, we conclude that the design of the direct-current power system meets the applicable regulations and is, therefore, acceptable. There is one area of the direct-current system which is presently under continuing post-Preliminary Design Approval review on the GESSAR-238 docket. The staff has requested that General Electric provide the independence requirements for the battery division which supplies the reactor protection system and engineered safety features system. The applicants have provided sufficient direct-current system capability but the area of the independence requirements has not been addressed because of the above cited continuing review. When resolution is achieved for GESSAR-238 in this area, the applicants have agreed to adopt the same requirements and design, since the direct-current system is not in the General Electric scope of supply for this application. We conclude this is acceptable since this will permit a firm design to be approved prior to a decision for issuance of construction permits for Montague 1 and 2. We will discuss approval of the design in a supplement to this report.

8.4 Physical Independence of Electrical Systems

The applicants have identified the recommendations of Regulatory Guide 1.75 as the primary design objectives for physical independence of the plant's electrical system. In addition, in those areas where tests or analyses are used to demonstrate conformance to the recommendations of the guide, the applicants have committed to specifically identifying the area and providing the results of the tests and/or analyses as part of the Final Safety Analysis Report.

We find the design criteria and the above commitment acceptable for the construction permit stage of our review.

8.5 Electrical Fire Prevention and Control

In response to a staff request regarding fire protection and control, the applicants provided a description of the plant's fire stops and seals and also a description of the fire detection and protection system and equipment.

The design criteria for the fire stops included the following:

(1) A fire rating consistent with the fire rating of the penetrated wall, floor, or ceiling.

(2) Suitability to penetration geometry and arrangement.

(3) Compatibility with cable and insulation materials.

(4) Ability to withstand maximum required pressure on either side of the penetration.

Although some of the design requirements are presently under development, the applicants have stated their intent to incorporate all applicable recommendations and requirements which result from present and ongoing studies.

The proposed design criteria and developmental program, with the commitment to incorporate the resulting study, recommendations and requirements into the design, are generally acceptable for the construction permit stage of review. However, since a decision on issuance of construction permits for Montague 1 and 2 will probably not be reached until at least 1979, we intend to keep this matter open for further in-depth review as the results of on-going studies develop. We will report on the final resolution of this matter in a supplement to this report.

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9.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown capability include the following: the standby service water system, portions of the reactor plant component cooling water system, the ultimate heat sink, the control building ventilation system, portions of the containment ventilation system, the filtration portion of the fuel building ventilation system, portions of the standby generator ventilation system, the standby generator and supporting systems, and the fire protection system.

The auxiliary systems necessary to assure the safe storage, handling, and cooling of the fuel include the fuel handling system, the new and spent fuel storage systems, the fuel pool cooling and cleanup system and the fuel building ventilation system.

We have reviewed other auxiliary systems whose failure would not prevent safe shutdown but could, either directly or indirectly, lead to potential release of radioactivity to the environment. These systems include the equipment and floor drainage system and the main steam isolation valve leakage control system.

We have also reviewed the design of those auxiliary systems whose failure would neither prevent safe shutdown nor result in potential radioactive release. These include non-safety portions of the component cooling water system, the makeup water treatment system, the condensate storage facilities, the plant service water system, the compressed air system, the process sampling system, the ventilation systems for non-safety related areas, and the communication and lighting systems. Failures of the above systems will not affect the capability of safety-related systems to effect safe shutdown. We conclude that the proposed designs for the above systems are acceptable.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel storage racks will provide for dry storage for approximately 30 percent of a full core load. The outer structure of the storage racks will be designed to preclude the inadvertent placement of a fuel assembly in the rack closer than the prescribed spacing. The storage racks will be designed so that the maximum effective multiplication (K_{eff}) will not exceed 0.95 in the event the new fuel area were flooded with water.

The new fuel storage racks will be bolted together and supported from the pool wall. The fuel racks will be designed to withstand the impact of a dropped fuel assembly. The new fuel racks will be designed to seismic Category I requirements.

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We have reviewed the adequacy of the applicants' proposed design criteria and design bases for the new fuel storage facility to assure maintenance of a subcritical array. We conclude that the design criteria and design bases are in conformance with General Design Criterion 62 and the staff positions of Regulatory Guide 1.13-Fuel Storage Facility Design Basis including the positions on seismic design and missile protection and are, therefore, acceptable.

9.1.2 Spent Fuel Storage

Spent fuel storage space will be provided in the fuel storage pool and in the containment upper fuel pool. The fuel storage pool will contain storage space sufficient for about 160 percent of a full core fuel load. The containment makeup pool will contain temporary storage space for an additional 25 percent of a full core. The spent fuel in both the fuel storage pool and the containment makeup pool will be covered with 33 feet of water. The spent fuel racks will be designed to provide protection against damage to the fuel and to prevent fuel assemblies from being stored in other than the prescribed locations. The maximum value for the effective multiplication of 0.95 will not be exceeded under any conditions.

The fuel pool will be designed to seismic Category I requirements. The facility will be designed to prevent the cask handling crane from traveling over or in the vicinity of the pool, thereby precluding damage to the stored spent fuel in the event of a dropped cask.

We have reviewed the adequacy of the applicants' proposed design criteria and design bases for the spent fuel storage facility to assure maintenance of a subcritical array during all normal, abnormal, and accident conditions. We conclude that the design of the spent fuel storage facilities will be in conformance with the requirements of General Design Criterion 62 and the positions of Regulatory Guide 1.13, including the positions on seismic design and missile protection, and in conformance with Branch Technical Position APCSB 9-1 with respect to crane travel over the pool and are, therefore, acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system is designed to remove the decay heat from the stored spent fuel elements, maintain purity and clarity of water in the spent fuel, cask storage pool and the fuel transfer pool. The spent fuel cooling system contains two seismic Category I trains which normally dissipate heat to the component cooling water system or the standby service water system and, under emergency conditions, to the residual heat removal system heat exchangers. Each train of the fuel pool cooling system will be designed to remove the decay heat generated by the normal storage load of the spent fuel in the pool and maintain the pool water temperature below 125 degrees Fahrenheit. The maximum possible heat load obtainable will be the decay heat of the full core load of fuel at the end of a fuel cycle plus the remaining decay heat of the spent fuel discharged from previous refuelings. The residual heat

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removal system in this case, may be operated in conjunction with the fuel pool cooling system to maintain the pool temperature below 125 degree Fahrenheit. Normally, the system will be isolated from the fuel pool cooling system by manually operated valves. Makeup water from the condensate storage tank will be provided to the spent fuel storage pool. Redundant trains of the seismic Category I standby service water system will be used as alternative sources of makeup water in case of failure of the normal makeup system.

We reviewed the adequacy of the applicants' proposed design criteria and design bases for the spent fuel pool cooling and cleanup system to assure continuous cooling during normal, abnormal, and accident conditions. We conclude that the design criteria and design bases are in conformance with General Design Criterion 44 and the positions of Regulatory Guide 1.13-Fuel Storage Facility Design Basis, including the positions on seismic design, missile protection and availability of assured makeup water systems and are, therefore, acceptable.

9.1.4 Fuel Handling System

The fuel handling system will be designed to safely handle fuel assemblies from receipt of new fuel to shipment of spent fuel. The system will be designed to conduct all spent fuel transfer and storage operations under water to ensure adequate shielding during refueling.

The arrangment of the fuel handling area will include a 125-ton overhead crane for the handling of the spent fuel shipping cask with a five-ton capacity auxiliary hoist. The design of the spent fuel shipping cask crane will be of the gantry type, seismic Category I. A five-ton crane and the general purpose grapple will be used to handle the fuel. An analysis was performed to evaluate the effects of a vertical or tipped drop of the spent fuel cask in the cask storage and washdown area. It was found that the travel of the spent fuel cask handling crane will be limited by physical arrangement from carrying the spent fuel cask over the spent fuel storage pool. The drawings also indicated that no safety-related equipment will be located near the cask storage compartment or near the spent fuel cask handling area, thus, no damage to spent fuel could result from an accidentally dropped cask.

We have reviewed the adequacy of the applicants' proposed design criteria and design bases to assure safe operation of the fuel handling system during normal, abnormal and accident conditions. We conclude that the design criteria and design bases are acceptable.

9.2 Water Systems

9.2.1 Station Service Water System

The station service water system for each unit will consist of both a normal service water system and a standby service water system. The applicants state that there will be no crossover of service water systems between units.

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The normal service water system is designed to provide cooling to both safety-related and non-safety related plant auxiliaries. The safety-related auxiliaries are the control room chillers, the residual heat removal heat exchangers during normal shutdown generators. The non-safety related auxiliaries are the turbine plant component cooling water heat exchangers, the reactor plant component cooling water heat exchangers, and the plant water chillers. The water source for the normal service water system is the cooling tower basins of the circulating water system. Make-up water will be supplied from the Connecticut River. The safety-related and non-safety related portions of the station service water system are separated by redundant automatic isolation valves on both the supply and return lines.

Each unit will have independent seismic Category I standby service water systems. These system combined with the ultimate heat sink will provide for the dissipation of residual heat from one unit during a plant shutdown while the second unit could be experiencing a design basis accident. The standby service water system will supply cooling water to the residual heat removal heat exchangers, standby generators, reactor plant component cooling water heat exchangers, and the control room air conditioning chillers. The standby service water systems will be a redundant system containing two 50 percent capacity pumps in each train. Each system will take its cooling water supply from one of two standby cooling towers and will discharge back to the same tower.

Electric power will be supplied from either offsite or separate and redundant onsite standby power sources. Either one of the standby service water pumps will be sufficient to handle the cooling requirements of any combination of components necessary to safely shutdown one unit following an accident, with the exception of the residual heat removal heat exchanger. When this exchanger is required the second standby service water pump will be used.

Based on our review, we find that the station service water system design criteria and bases are in conformance with General Design Criteria 44, 45, and 46. We conclude, therefore, that the proposed system is acceptable.

9.2.2 Reactor Plant Component Cooling Water System

The reactor plant component cooling water system provides cooling water to reactor auxiliary components during all normal modes of operation and during a loss of offsite power. During faulted conditions, standby service water will be provided by seismic Category I designed connections to those components of the system that are essential. Automatic valves in series with check valves on the inlet side with double automatic valves on the return side to the standy service water system will isolate the safety-related seismic Category I components from the non-safety related components.

This system is a closed loop consisting of three 50 percent capacity pumps and three 50 percent capacity heat exchangers. During normal operation, only two component cooling water system pumps and two component cooling heat exchangers will be required to handle the heat removal loads. The third pump and third heat exchanger will be on standby and will provide for redundancy. During normal conditions makeup water will be supplied from the condensate makeup and drawoff system to the expansion tank. Emergency makeup water will be provided from the seismic Category I portion of the service water system.

The safety-related portions of this system will either have full capacity redundant components or be designed so that any two of three components will be available to perform their functions. The system will be monitored for radioactive inleakage.

Based on our review, we conclude that the reactor plant component cooling water design criteria and bases are in conformance with the requirements of General Design Criterion 44 regarding the ability to transfer heat from safety-related components to the ultimate heat sink under normal and accident conditions. We further conclude that the system design criteria and bases meet the requirements of General Design Criteria 45 and 46 in regard to system design for periodic inspections and tests, including functional testing and confirmation of heat transfer capabilities. We, therefore, conclude that the proposed system is acceptable.

9.2.3 Ultimate Heat Sink

The ultimate heat sink will consist of two 100 percent capacity cooling towers and storage facilities. The system will be designed to seismic Category I requirements and in accordance with Position 2 of Regulatory Guide 1.27 - Ultimate Heat Sink for Nuclear Power Plants. The cooling tower fans will be mounted in a stack designed to withstand tornado missiles in accordance with requirements in General Design Criterion 4. In the event of a loss of offsite power, the power required to operate each set of two standby cooling tower fans will be supplied by one of four site standby generators, in accordance with requirements in General Design Criterion 44. The water inventory lost due to natural evaporation during normal plant operating conditions will be made up to each basin from the site normal makeup water system. The applicants' analysis of the water inventory is based on the assumption that one unit experiences a loss-of-coolant accident while the other unit is undergoing normal shutdown and cooldown. The standby service water system will be recirculated to the ultimate heat sink for a period of 30 days, assuming no makeup water is available. We have reviewed this analysis and find that the water in storage will be sufficient to assure that evaporation and wind drift loss during the 30-day period will not reduce the water inventory of the basin to an unacceptable level.

We have reviewed the applicants' proposed design crite[illegible] design bases for the ultimate heat sink necessary to dissipate heat under [illegible]dent conditions. We conclude that the design criteria and design bases [illegible] guidance of the positions of Regulatory Guide 1.27 and, therefore, find t[illegible]ystem acceptable.

9.2.4 Plant Chilled Water System

The plant chilled water system will be designed to provide 44 degrees Fahrenheit chilled water to cooling coils required to dissipate heat loads from equipment

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located in confined areas. The cooling coils to be located in the containment, auxiliary, and fuel building will be safety-related but the coil located in the turbine building will not.

As a result of our review and, based on the applicants' proposed design requiring that: (1) piping that connects to the cooling coils that are safety-related will be designed to seismic Category I criteria, in accordance with requirements in General Design Criterion 2, and will be supplied with water from the standby service water system; and (2) double automatic isolation valves will be provided on the supply and return piping between the non-safety related and safety-related portions of the plant chilled water system, in accordance with requirements in General Design Criterion 5, we find the system acceptable.

9.2.5 Control Building Chilled Water System

The control building chilled water system will be designed to provide 44 degrees Fahrenheit chilled water to the cooling coils used for space cooling and dehumidification. The system will be required to operate during normal shutdown and post-accident conditions without loss of any safety-related function.

The system will be fully redundant and designed to seismic Category I requirements. The system will consist of three 100 percent capacity water chillers, three chilled water circulation pumps, three condenser water recirculation pumps, and two expansion tanks.

The condensers will be connected to the standby service water system to ensure that the system safety function is met in the event of loss of normal service water system water supply. All the equipment associated with the system will be located in the control building, a tornado protected, seismic Category I structure.

Based on our review, we conclude that the design criteria and bases for the chilled water system of the control building are in conformance with General Design Criteria 2, 5, and 44 and are, therefore, acceptable.

9.3 Process Auxiliaries

9.3.1 Equipment and Floor Drainage System

The equipment and floor drainage system will accommodate drains from the containment building, the auxiliary building, the fuel building, the radwaste building, and the turbine building. Drainage in these buildings will collect in sumps before being transferred to the appropriate system. Drains from potentially radioactive sources will be routed to either the main condenser hotwell or the radioactive liquid waste system depending on the conductivity of the water.

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Drains from areas or compartments containing engineered safety feature equipment will be separated from the remainder of the system. All sumps will contain two full size pumps. Level switches will provide automatic sump pump operation and level alarm.

All emergency core cooling system equipment and reactor core isolation coolant system cubicles that are located in the lowest level of the auxiliary building will be watertight compartments, serviced by their own independent floor drainage sumps and pumps. Each sump will house duplex pumps and be of sufficient volume to handle normal leakage. Since there will be no openings at the lower level of the cubicles and cubicle piping penetrations will be sealed, the cubicles cannot be flooded if the surrounding area is flooded. Redundant engineered safety feature equipment will be located in separate cubicles to ensure safe shutdown capability should one system become inoperative.

Safety-related equipment in the fuel building, the standby generator building and the control building will be protected from flooding by watertight compartments. We find this acceptable.

Based on of our review, we find that the system is adequately designed to prevent flooding in areas containing safety-related equipment and, therefore, we conclude that the system is acceptable.

9.3.2 Main Steam Line Isolation Valve Leakage Control System

The applicants in Section 9.3.6 of the application have committed to provide a main steam line isolation valve leakage control system which will be designed in accordance with the guidelines in Regulatory Guide 1.96 - Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Nuclear Reactors. Since the preliminary design of this system has not been completed, we requested and the applicants agreed to design this leakage control system in accordance with the Regulatory Guide 1.96 and any revisions issued up to the time the design is submitted for our review. Based on the present availability of an approved leakage control system for boiling water reactors, we conclude the applicants' commitment to provide a system designed in accordance with Regulatory Guide 1.96 is acceptable.

9.4 Heating, Ventilation and Air Conditioning Systems

The engineered safety feature ventilation system is required to maintain a controlled environment in areas containing equipment that must remain operable during a design basis accident and continue to function during post-accident conditions. To assure its availability, these systems will be designed to seismic Category I criteria, will be powered from emergency buses, and will be supplied with cooling water from the standby service water system. The following locations will contain engineered safety feature equipment: the standby service water pump house; the diesel generator rooms; the emergency core cooling system pump rooms; the fuel building; the control building; and the containment. The standby service water pump house will be cooled by power operated roof vents and will be heated by electric space heaters. The above engineered safety feature equipment ventilation systems are discussed below except for the containment ventilation system which is discussed in Section 6.2.3 and 6.2.4.

9.4.1 Control Building System

The control building heating, ventilation and air conditioning system will be designed to maintain the control room, electrical auxiliary rooms, cable spreading room, battery rooms, and engineered safety feature heating, ventilation and air conditioning equipment room, within the thermal and air quality limits required for operation of plant controls and uninterrupted safe occupancy of those areas that have to be occupied during normal operation, shutdown and post-accident conditions.

The control room system will consist of a normal air-conditioning system and a standby cooling and filtering system. During accident conditions, the control room and the essential standby system will be automatically isolated from the normal non-seismic Category I system by redundant seismic Category I isolation valves.

The standby control room system and those portions of the control building system that are used to maintain the cable spreading rooms, switchgear rooms, battery rooms and the engineered safety feature heating, ventilation and air conditioning equipment room within environmental design limits will be designed to seismic Category I requirements and each will consist of two independent 100 percent capacity systems.

These systems will be designed to maintain the control room and the balance of the building under positive pressure. Missile protected, outside air intakes will be provided for the standby and for the normal systems. Redundant smoke and radiation detectors will monitor the outside air supply with alarms in the control room. The radiation detector will also automatically isolate the outside air supply and start the standby heating, ventilation and air conditioning and air filtration system.

We have reviewed the proposed design criteria and bases and find that the proposed system design meets the requirements set forth in General Design Criterion 19, in regard to the capability for operating the plant from the control room during normal and accident conditions. Therefore, we conclude that the system is acceptable.

9.4.2 Auxiliary Building Ventilation

The auxiliary building ventilation system will be designed to provide normal ventilation for the emergency core cooling system pump rooms and all other areas in the auxiliary building and maintain a negative pressure in the auxiliary building, including the emergency core cooling system pump rooms.

Two seismic Category I isolation dampers will be installed in series to isolate the normal outside air intake ducts and the exhaust ducts for this system. These isolation dampers will be closed upon detection of high radiation in the area. The

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exhaust air subsystem will function during all operating modes including accident conditions. The system will consist of two 100 percent capacity exhaust fans. One fan will be on standby and will start automatically upon failure of the operating fan. The exhaust air flow will be monitored continuously for radioactivity by two detectors which will automatically, upon detection of high radiation level or during a loss-of-coolant accident, close the supply air dampers, shutdown the supply fans and divert the exhaust air to the standby gas treatment system thus automatically isolating the auxiliary building. During accident conditions, the building air supply fans will be automatically stopped, intake dampers will be closed and the exhaust fans started by the standby electrical power sources.

The system will include a fan coil unit for each emergency core cooling system equipment area. Each fan coil unit will be capable of maintaining the room temperature below 150 degrees Fahrenheit. The fan coil will be powered from the emergency bus serving the associated emergency core cooling system equipment. The cooling coils will be supplied with water from the control building chilled water system which in turn will be cooled by the standby service water system. The fan coil units will be designed to meet seismic Category I requirements.

Based on cur review and evaluation of the applicants' proposed design criteria and design bases for maintenance of a suitable environment for essential equipment and to preclude the unacceptable release of contaminants to the environment during normal, abnormal, and accident conditions, we conclude that the system is acceptable.

9.4.3 Fuel Building Ventilation

The ventilation system for the fuel building will be designed to maintain the building space temperature between a minimum of 60 degrees Fahrenheit and a maximum of 104 degrees Fahrenheit. The supply air subsystem will be designed to function during normal operating conditions only. The exhaust air subsystem and the recirculation air subsystems will function during normal operating and accident conditions. In the event of a fuel handling accident or if high radiation is detected in the fuel building, exhaust air will be automatically diverted through a charcoal filter train.

The exhaust air subsystem will include two fully redundant charcoal filter trains, and two 100 percent capacity exhaust fans, one operating during normal operation and one on standby. The exhaust air fans will be connected to the normal station service power supply and to independent standby buses. During normal operation the exhaust air will be monitored continuously. During spent fuel handling operations, exhaust air will be diverted through one of the charcoal filter trains as an administrative procedure, thus ensuring no radioactivity releases in the event of a fuel handling accident.

The recirculation air subsystem will be designed to seismic Category I criteria, and will provide the principal cooling mode for all operating and accident conditions. This subsystem will reduce the exhaust air flow rate from the building after a fuel handling accident to ensure that the release of radioactivity will be maintained within acceptable limits. Two seismic Category I isolation dampers will be installed in series to isolate the normal outside intake duct for the fuel building ventilation system.

Based on our review of the fuel building ventilation system, we conclude that the proposed system is acceptable.

9.4.4 Standby Generator Room Ventilation

The standby generator room ventilation system will be designed to ventilate and to maintain room temperature between 50 and 104 degrees Fahrenheit. The portions of the system whose function is to control the room temperature, are safety-related and will be designed to seismic Category I criteria. Each standby generator room will have its own ventilation system which will operate independently of the others.

Each subsystem for maintaining room temperature will consist of a missile protected air intake opening; a 100 percent capacity air supply fan; supply, recirculation, and exhaust ductwork; dampers, and a missile protected exhaust opening. Each fan will be powered by the normal station power supply and the standby generator that it serves.

The combustion air supply and exhaust systems are not a part of the diesel generator room ventilation system. Each standby generator will be provided with separate combustion air supply and engine exhaust systems which will be designed to seismic Category I criteria. The diesel engine exhaust will be located approximately 10 feet above the roof and approximately 50 feet away from the combustion air intake, thus assuring that the exhaust gases will not affect the quality of the combustion air.

Based on our review, we conclude that the system capacity and design criteria can satisfy their designated safety function and are, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The fire protection system will provide fire protection capability in those areas of the plant where a fire hazard may exist. This system will be designed to: (1) provide a reliable and adequate supply of water to meet any probable demand with a sufficient number of strategically located yard fire hydrants and small, fire hose connections in the areas of fire potential throughout the plant; (2) provide portable fire extinguishers of the proper types throughout all plant areas; (3) provide fixed automatic sprinkler or deluge systems in areas of fire potential greater than those that can be extinguished with portable or manual equipment; (4) provide fire and smoke detection and monitoring systems for the switchgear area, battery room, control room, cable penetration area and all other areas where the danger of the fire exists; and (5) provide chemical extinguishing systems where automatic sprinkler or deluge systems are not appropriate. Non-toxic gaseous extinguishing materials will be used in areas normally occupied by station personnel. The system will be designed to comply with the Standards of the National Fire Protection Association.

The plant design will emphasize fire prevention by using non-combustible or fire resistant materials to the greatest extent possible. The integrity of vital areas, components and systems will be assured through redundancy, physical separation, and engineered fire barriers.

Water for fire protection will be supplied from two fire water storage tanks which will receive makeup from the Connecticut River. This system contains two fire pumps, each with a capacity of 2500 gallons per minute (one diesel engine driven, the other electric motor driven). An underground yard water loop will surround the entire station and provide water to hydrants, the interior fire hose piping and automatic sprinkler or deluge system. Valves will be provided for isolating portions of the system when required.

The penetration and isolation valves for the line supplying water to the reactor building hose stations of the system will be designed to seismic Category I criteria. All other portions of the system will be non-seismic Category I. However, all fire protection piping containing water in the vicinity of any engineered safety feature equipment will be seismically analyzed to ensure that allowable design stresses will not be exceeded.

Air foam fire protection systems will be provided to protect the condenser pit areas and the fuel oil tanks. These systems will be manually actuated.

A flooding carbon dioxide fire protection system will be provided for the following area: (1) normal switchgear rooms; (2) the two standby switchgear rooms; (3) the high pressure core spray switchgear room; (4) the cable chase areas, (5) the exciter housing of the main generator; and (6) the relay and computer areas. Actuation of the fixed carbon dioxide system will be automatic except in the computer room where the system will be manually actuated.

The control room in the control building will be equipped with hand-operated portable carbon dioxide and pressurized water fire-extinguishers and portable breathing apparatus. A fire outside the control room will not preclude continued control room habitability because multiple control room air intakes with smoke detectors are provided. In the unlikely event that a fire within the control room necessitates the evacuation of the control room, the reactor can be safely shutdown from a remote location.

Smoke detectors will be monitored on an annunciator panel in the control room to alert personnel of a possible fire situation in the following areas: normal and standby switchgear rooms, cable chase areas, relay room, computer room, cable tunnel in turbine building, control building air conditioning equipment room, containment at strategic cable areas and heating, ventilation and air conditioning equipment rooms.

In summary, based on our review to date the overall fire protection system for Montague 1 and 2, we conclude that the design criteria and bases meet the requirements of General Design Criterion 3 and therefore, forms an adequate basis for acceptance. However, as a result of generic investigations and guidelines being developed by the staff on fire protection systems, further requirements may be imposed on Montague 1 and 2 to further improve the capability of the proposed fire protection system to prevent unacceptable damage that may result from a fire. Any additional requirements or modifications resulting from the above studies will be reported in a supplement to this report.

9.5.2 Standby Generator Fuel Oil Storage and Transfer Systems

The standby generator fuel oil storage and transfer systems will be designed to provide sufficient storage of fuel oil to allow operation of each emergency diesel generator for a minimum of seven days. The system for each unit will be designed to seismic Category I requirements and consist of three independent trains, one for each diesel generator. Each train will include a storage tank and two pumps. The entire fuel oil storage and transfer systems, except for the storage tanks which are underground, will be located in a tornado protected structure.

The systems will be designed to perform their function when required for the following conditions: loss of offsite power coincident with failure of one standby generator; loss of offsite power coincident with maintenance outage or failure of one fuel oil transfer pump associated with each standby generator.

Based on our review, we conclude that the capacity of the systems and the design criteria satisfy the requirements of their designated safety function and are therefore acceptable.

9.5.3 Generator Auxiliary Systems

The diesel generator auxiliary systems will include the diesel generator cooling water system, diesel generator starting system, and the diesel generator lubrication system.

The diesel generator cooling water system will be designed to maintain the temperature of the diesel engine within a safe operating range. This system will be a closed cooling system and the heat will be rejected to the service water system. When the engine is idle, the engine water will be heated by immersion heaters to keep the engine warm and ready to start and accept loads within the prescribed time interval. The system will be designed to seismic Category I requirements.

Each diesel generator will be provided with two separate and independent compressed air starting trains consisting of an air compressor and starting air storage tank. Each tank will be capable of providing five starts without recharging from the diesel generator compressors. Except for the compressors, the starting air system will be designed to the requirements of seismic Category I criteria.

Each diesel generator will be provided with a lubrication system designed to supply lubricating oil to the diesel generator system. The system will circulate lube oil through the engine for heating when the engine is idle and for cooling when the engine is operating. The system will be designed to seismic Category I criteria.

Based on our review, we conclude that the design criteria and bases meet the requirement of their designated safety functions, have the needed capacity and are therefore acceptable.

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10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Turbine Generator

The turbine electro-hydraulic control system will control the speed of the turbine (1800 revolutions per minute, rated) by modulating the turbine inlet steam control valves to regulate the steam flow to the turbine.

The turbine control system will be designed to trip the turbine under the following conditions: turbine overspeed, loss of condenser vacuum, excessive thrust bearing wear, reactor trip, generator electric trip, loss of hydraulic fluid supply pressure, low bearing oil pressure and manual trip from the control room or at the turbine.

The turbine generator will be provided with two independent overspeed protection systems, an electrical overspeed trip device and a mechanical overspeed trip device. The redundant overspeed trip devices will trip the turbine at approximately 110 percent of turbine rated speed by closing all the turbine inlet valves. Based on the design of and the redundancy in the turbine overspeed protection system, we conclude that the turbine will be protected from excessive overspeed.

We reviewed the adequacy of the applicants' proposed design criteria and design bases to assure safe operation of the turbine generator under normal, abnormal and accident conditions. Based on this review, we conclude that the design criteria and design bases are acceptable.

10.2 Turbine Missiles

The applicants have arranged the two turbine units in a peninsular orientation with respect to their respective reactor containment buildings. All seismic Category I structures will have concrete exterior walls with a minimum thickness of 30 inches. In addition, the turbine generators will be flanked on either side by one-foot thick concrete cubicles which house the moisture separator reheater units.

We find that the proposed plant design, with respect to potential turbine missiles, is acceptable.

10.3 Main Steam Supply System

The steam generated in the reactor will be routed to the high pressure turbine by means of four main steam lines. Each main steam line will contain two main steam isolation valves and one shutoff valve. The portion of the main steam supply system from the reactor through the outermost containment isolation valve to the main steam shutoff valve will be designed to seismic Category I criteria.

The main steam isolation valves will be designed to close on loss of pneumatic pressure to the valve operator and will close within 5.5 seconds, including the actuation instrument delay following a condition requiring isolation. An accumulator for each isolation valve will provide pneumatic pressure for valve closure in the event of a failure of the normal air supply system. The valves will be designed to close for the condition of the maximum mass flow rate in either direction in the event of a double-ended steam line break. Redundancy of isolation valves in the main steam lines satisfy the single failure criterion and will prevent complete blowdown of the reactor system in the event of a steam line break outside the primary containment.

Based on our review, we conclude that the main steam supply system design criteria and bases are in conformance with the single failure criterion, the seismic design position of Regulatory Guide 1.29 - Seismic Design Classification, and the valve closure time requirements and are, therefore, acceptable.

10.4 Other Features of Steam and Power Conversion System

10.4.1 Circulating Water System

The circulating water system will furnish the main steam condenser with cooling water from the natural draft cooling tower at a total design flow rate of 625,000 gallons per minute. We have reviewed the consequences (flooding) resulting from a failure of this system in terms of the effects on the safety-related equipment of the plant. A circulating water line expansion joint rupture may result in pump runout. The potential of failure of the expansion joints is minimized by designing the joints to withstand the pump shut off head. The condenser will be connected to the circulating water piping using expansion joints located between the condenser and the motor operated butterfly valve on both sides of the condenser. An enclosure will be built around the condenser expansion joints to contain and minimize leakage, should an expansion joint fail. The enclosures will be sized so that expansion joint failure will not result in flooding of the area. One level switch with an alarm in the control room will be provided for each expansion joint protective enclosure to alert the operator of excessive leakage or expansion joint failure. Limit switches will be provided on the water box butterfly valves to minimize the possibility of expansion joint rupture due to pressure surge resulting from pumping against a closed valve. A failure in the circulating water system or the condensate system large enough to cause flooding will be detected by high level alarms in the turbine room sumps and condenser pits. The alarm will alert the operator to take action in isolating the equipment or shutdown of the system. There will be no safety-related equipment in the turbine building that can be affected by flooding. Makeup water to offset cooling tower evaporation, drift losses and blowdown will be pumped from the Connecticut River. The system will be designed to non-seismic Category I criteria.

We reviewed the adequacy of the applicants' proposed design criteria and design bases to assure safe operation of the circulating water system during normal, abnormal, and accident conditions. Based on this review, we conclude that the design criteria and design bases are acceptable.

10.4.2 Condensate Demineralizer System

The condensate demineralizer system will be designed to maintain the condensate at the required purity by removal of contaminants. The system will be capable of purifying condensate up to a maximum flow rate of approximately 12,300,000 pounds per hour. The system will be designed to provide "reaction" time to take corrective action or initiate a unit shutdown in the event of massive leaks, such as complete failure of a condenser tube. Standby equipment will be provided to preclude difficulties in handling radioactive waste when the system is operating at normal influent concentrations. An effluent strainer in the piping from each ion exchanger will protect the system against massive discharge of resin in the event of an underdrain failure. Conductivity alarms will be provided to alert the operator to off-normal conditions and the resin condition will be monitored in accordance with the provisions of Regulatory Guide 1.56 - Maintenance of Water Purity in Boiling Water Reactors.

Based on our review, we conclude that the condensate demineralizer system will be designed to adequately perform its function and is acceptable.

10.4.3 Turbine Disk and Rotor Integrity

The probability of failure of a turbine disk or rotor at speeds up to design overspeed can be minimized by the use of suitable materials, adequate design, preservice spin tests, and preservice and inservice inspections. The applicants have described a program for assuring the integrity of turbine disks and rotors by the use of suitable materials with adequate fracture toughness, suitable design practices, preservice spin tests, and preservice and inservice inspections. We have reviewed the proposed provisions, and conclude that they provide reasonable assurance that the turbine disks and rotors will not fail during normal operation, including transients up to five percent above the anticipated speed resulting from a loss of load. We find this to be acceptable.

10.4.4 Oteam and Feedwater System Materials

The mechanical properties of materials selected for Class 2 and 3 components of the steam and feedwater systems will satisfy Appendix I of Section III of the ASME

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Boiler and Presure Vessel Code and Parts A, B, and C of Section II of the Code. The fracture toughness properties of the ferritic materials will satisfy the requirements of Articles NC-2300 and ND-2300 of Section III, 1974 Edition of the ASME Code.

The controls that will be imposed upon the austenitic stainless steel are in conformance with the provisions of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and 1.44, "Control of the Use of Sensitized Stainless Steel." The applicants have agreed to demonstrate the adequacy of current welding controls by conducting tests to determine the ferrite content of production welds and to evaluate the degree of sensitization in welded type 304 and 316 stainless steel. Fabrication and heat treatment practices performed in accordance with these requirements will provide added assurance that stress corrosion cracking will not occur during the design life of the plant.

Conformance with the Codes and Regulatory Guide recommendations cited above constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting the requirements of General Design Criterion 1.

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

Units 1 and 2 will share the liquid and solid radioactive waste management systems and will have separate and identical gaseous radioactive waste management systems. These systems will be designed to provide for controlled handling and treatment of liquid, gaseous and solid wastes. The liquid radwaste system will process liquid wastes from equipment and floor drains, coolant leakage, condensate demineralizer regenerant liquids, ultrasonic resin cleaning wastes and decontamination and laboratory wastes. The gaseous radwaste system will provide holdup capacity to allow for decay of short-lived noble gases that are taken from the main condenser offgas system and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers. The radwaste systems will be designed to reduce releases of radioactive materials in effluents to the as low as reasonably achievable levels in accordance with Section 50.34a of 10 CFR Part 50. The solid waste system will provide for the solidification, packaging and storage of solid radioactive wastes generated during station operation prior to their shipment offsite for burial. Solid packaged wastes will be shipped to a licensed facility for burial.

In our evaluation of the liquid and gaseous radwaste systems, we considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents as low as reasonably achievable, based on expected radwaste inputs over the life of the plant; (2) the capability of the systems to maintain releases below the limits in Section 20.106 of 10 CFR Part 20, during periods of fission product leakage at design levels from the fuel; (3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences; (4) the quality group and seismic design classification applied to the system design; (5) the design features that will be incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60; and (6) the potential for gaseous release due to hydrogen explosions in the main condenser offgas treatment system.

In our evaluation of the solid radwaste treatment system, we considered: (1) system design objectives in terms of expected types, volumes and activities of wastes processed for offsite shipment; (2) waste packaging and its conformance to applicable Federal packaging regulations; (3) provisions for controlling potentially radioactive airborne dusts during baling operations; and (4) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent radiological monitoring and sampling systems, we considered the systems' capability to; (1) monitor all normal and potential pathways for release of radioactive materials to the environment; (2) control the release of radioactive materials to the environment; and (3) monitor the performance of process equipment and detect radioactive material leakage between systems.

In the Final Environmental Statement for Montague 1 and 2, scheduled to be issued in August 1976, we will report on our evaluation to determine the quantities and activities of materials that will be released in liquid and gaseous wastes, or shipped offsite as solid wastes for burial. In that evaluation, we considered waste flows, waste activities, and equipment operating performance, including anticipated operational occurrences, that are consistent with an assumed 30 years of normal plant operation. The liquid and gaseous source terms listed in Tables 3.3 and 3.4 of the Final Environmental Statement were calculated using the BWR-GALE Code described in Draft Regulatory Guide 1.CC, "Calculations of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors." The principal parameters used in these calculations, along with their bases, are given in Appendix B to Draft Regulatory Guide 1.CC.

Based on the following evaluation we conclude that the liquid, gaseous and solid radwaste treatment systems and associated process and effluent radiological monitoring and sampling systems are acceptable and that the effluents concentrations will meet as low as reasonably achievable levels in accordance with 10 CFR 50.34a, Section II.A, II.B and II.C of Appendix I to 10 CFR Part 50, and the alternative to Section II.D of Appendix I as provided in the Annex to Appendix I as amended (September 4, 1975). The applicants have chosen the alternative provided in the Annex and, therefore, no cost-benefit analysis pursuant to Paragraph II.D. has been performed.

11.2 System Description and Evaluation

11.2.1 Liquid Radioactive Waste Treatment System

The liquid radioactive waste treatment system will consist of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of liquid radioactive wastes. The liquid radioactive wastes will be processed on a batch basis to permit optimum control of the disposition of processed wastes. Prior to being recycled or released, samples of processed wastes will be analyzed to determine the types and amounts of radioactivity present. Based on the results of the analysis, the wastes will be retained for further processing, recycled for eventual reuse in the plant, or released under controlled conditions. Liquid radioactive wastes will be segregated based on their origin and processed through either the waste collector subsystem, the floor-drain subsystem, the regenerant chemical subsystem or the phase separator subsystem. All laundry will be shipped offsite to a commercial laundry, and there will be no laundry facilities at the plant site. The principal components making up each of these subsystems, along with their principal design parameters, are listed in Table 11.2.1 of this report.

The design capacities of the waste collector subsystem demineralizers will be 20,000 gallons per day. The design capacities of both the floor drain subsystem and regenerant chemical subsystem evaporators will be 58,000 gallons per day. There is an interconnection between the floor drain subsystem and regenerant chemical subsystem so that the evaporator in each system will provide the redundancy to the other. We calculated the average expected waste flows to the waste collector, floor drain and regenerant chemical subsystems to be 29,300, 6,300 and 1,700 gallons per day, respectively,

TABLE 11.2.1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS
CONSIDERED 'N RADWASTE SYSTEM EVALUATION

Components	No.	Capacity Each	Quality Group (1)
Main Condenser Offgas Treatment System			
Sacrificial Adsorption Beds	2	-	C
Catalytic Recombiner	2	-	D(Augmented)
Charcoal Tanks	8	7.5 tons	C
Other Components Upstream of Sacrificial Adsorption Beds	-	-	D(Augmented)
Other Components Downstream of Sacrificial Adsorption Beds	-	-	C
Liquid Radwaste System			
Waste Collector Tanks	6	25,000 gallons	D(Augmented)
Floor Drain Tanks	4	25,000 gallons	D(Augmented)
Regenerant Neutralizer Tanks	4	25,000 gallons	D(Augmented)
Waste Sample Discharge Tanks	3	25,000 gallons	D(Augmented)
Phase Separator Tanks	4	5,800 gallons	D(Augmented)
Recovery Sample Tanks	4	25,000 gallons	D(Augmented)
Evaporator Bottoms Tanks	1	300 gallons	D(Augmented)
Radwaste Demineralizers	6	200 gallons per minute	D(Augmented)
Waste Evaporator	1	40 gallons per minute	D(Augmented)
Regenerant Evaporator	1	40 gallons per minute	D(Augmented)
Solid Radwaste System			
Filter Sludge Holding Tank	1	1,500 gallons	D(Augmented)
Waste Sludge Tank	1	5,800 gallons	D(Augmented)

(1) Quality Group C components will be of seismic Category I design and Quality Group D components will be of non-seismic design.

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from each reactor. The difference between the expected flows and design capacity will provide adequate reserve for processing surge flows. We find that the system capacity and system design are adequate for meeting the demands of the station during anticipated operational occurrences.

The liquid radwaste system will be located in a seismic Category I structure. The seismic and quality group classifications of the equipment, which are consistent with our guidelines, are listed in Table 11.2.1. The systems will also be designed to control the release of radioactive materials due to overflows from tanks outside containment by providing level instrumentation which will alarm in the control room, and by means of curbs and retention walls to collect liquid spillage and retain it for processing in the liquid radwaste system. We find that these provisions will be capable of preventing the uncontrolled release of radioactive materials to the environment. We find the applicants' proposed system design to be in accordance with Branch Technical Position Effluent Treatment Systems Branch 11-1, and to be acceptable.

We have determined that during normal operation, the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.21 curies per year per reactor, excluding tritium and dissolved gases, and 28 curies per year per reactor for tritium.

Based on our evaluation, the release of radioactive materials in liquid effluents will not result in an estimated annual total body dose or dose commitment for any individual in an unrestricted area greater than 3 millirem per year or critical organ doses greater than 10 millirem per year, in accordance with Section II.A of Appendix I to 10 CFR Part 50.

The estimate of radioactivity in liquid effluents from the site, exclusive of tritium and dissolved noble gases, will be less than 5 curies per year. The total body and critical organ doses have been estimated to be less than 5 millirem per year from the site. This meets the requirements of the alternative to Section II.D of Appendix I for not performing the cost-benefit analysis as provided in the Annex to Appendix I.

11.2.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste treatment system will be designed to process gaseous plant wastes based on the origin of the wastes in the plant and the expected levels of radioactivity. The gaseous waste treatment system will consist of a process offgas system and building ventilation systems that will control the release of radioactive materials in effluents to the environment. The principal components of the system, along with their principal design parameters, are listed in Table 11.2.1.

This system will be designed to collect and delay fission product noble gases removed from the condenser by the air ejectors at the condenser design air in-leakage rate. In the process offgas system, the gas flows through a hydrogen recombiner, a condenser, a moisture separator, a sacrificial adsorption bed and a dryer. This portion of the system will consist of two separate trains of equipment which provides 100 percent redundancy in the processing of the gaseous wastes. From the dryers, the gaseous wastes will flow into a set of eight charcoal beds with a total of 60 tons of activated charcoal arranged in two identical trains of four beds each and refrigerated to zero degrees Fahrenheit. The trains can be operated in series or in parallel. In addition, either train may be isolated while the other handles full flow. Operating in this manner, the system will have adequate capacity to allow operation during periods of equipment downtime.

We find that the system capacity and the system design will be adequate for meeting the demands of the station during normal operations including anticipated operational occurrences.

The system design includes hydrogen analyzers upstream and downstream of the recombiners which will initiate an alarm so that appropriate action can be taken to ensure that hydrogen concentrations are maintained well below the flammable limits. The input stream to the recombiners will be diluted with steam to maintain hydrogen concentrations at less than four percent. The system will be designed to maintain its integrity in the event of a hydrogen explosion.

The portions of the system upstream of the isolation valve preceding the sacrificial charcoal beds will be designed to Quality Group D (augmented) and non-seismic Category I, and will be located in a non-seismic Category I structure. The remainder of the system, which will include the charcoal adsorber tanks, will be designed to Quality Group C, seismic Category I, and will be located in a seismic Category I structure. We find the system quality group and seismic design classification, and the design provisions incorporated to reduce the potential for hydrogen explosion to be acceptable.

The plant ventilation systems will be designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination. Ventilation exhausts from the fuel building, including the radwaste and fuel handling areas, will be processed through high efficiency particulate air filters and charcoal adsorbers prior to release. Ventilation air from the auxiliary building will normally be released without treatment. The applicants propose to use the standby gas treatment system to treat effluents from the auxiliary building in the event radiation measurements exceed a predetermined level. We find the applicants' proposed design to be acceptable.

Ventilation air from the radwaste building will be processed through high efficiency particulate air filters prior to release. Ventilation exhaust from the containment building and from the drywell will be processed through high efficiency particulate air filters and charcoal adsorbers prior to release. Ventilation exhaust air from the turbine building will be released without treatment.

We have determined that the proposed gaseous radwaste treatment systems and plant ventilation systems will be capable of reducing the release of radioactive materials in

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gaseous effluents to approximately 4,000 curies per year per reactor of noble gases and 0.31 curies per year per reactor of iodine-131, 68 curies per year per reactor of tritium, 9.5 curies per year per reactor of carbon-14, 25 curies per year per reactor of argon-41 and 0.036 curies per year per reactor of particulates.

Based on our evaluation, the release of radioactive materials in gaseous effluents will not result in an estimated annual air dose greater than 10 millirads per year for gamma radiation, 20 millirads per year for beta radiation, or a dose greater than 15 millirads per year to any organ for radioiodine and radioactive particulates in accordance with Sections II.B and II.C of Appendix I to 10 CFR Part 50.

The effluents from the site will not result in an annual gamma air dose greater than 10 millirads, a beta air dose greater than 20 millirads, a release of iodine-131 greater than 1 curie (per reactor), or a dose to any organ from radioiodine and radioactive particulates released greater than 15 millirem, in accordance with the alternative to Section II.D of Appendix I as provided in the Annex to Appendix I.

11.2.3 Solid Radwaste Treatment System

The solid radwaste treatment system will be designed to collect and process wastes based on their physical form and need for solidification prior to packaging. "Wet" solid wastes, consisting of spent demineralizer resins, evaporator bottoms, filter sludges, and filter demineralizer backwash, will be combined with a solidifying agent and a catalyst to form a solid matrix and will be sealed in shipping containers. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon steel drums. Miscellaneous solid wastes, such as irradiated primary system components will be handled on a case-by-case basis based on their size and activity. Expected solid waste volumes and activities shipped offsite for each reactor will be 42,000 cubic feet per year of "wet" solid waste containing an average of 0.1 curie per cubic feet and 4400 cubic feet per year of "dry" solid waste containing less than five curies of activity.

Container filling operations will be controlled remotely from consoles located outside the container fill area. Filling operations will have interlock features to prevent overfilling of a container. The dry waste compactor will be vented directly to the radwaste building vent.

The solid radwaste systems will be located in a seismic Category I structure. The seismic design and quality group classifications of the equipment, which are consistent with our guidelines, are listed in Table 11.2.1.

Storage facilities for up to 25 containers (50 cubic feet, each) of solid radioactive wastes will be provided at grade level in the radwaste building. Based on our estimate of 42,000 cubic feet per year per reactor, we find the storage capacity adequate for meeting the demands of the station. Wastes will be packaged in 50 cubic feet containers and in 55-gallon steel drums in accordance with the requirements of 10 CFR Part 20, 10 CFR Part 71 and 49 CFR Parts 170-178, and shipped to a licensed burial site in accordance with NRC and Department of Transportation regulations.

11.3 Process and Effluent Radiclogical Monitoring and Sampling Systems

The process and effluent radiological monitoring and sampling systems will be designed to provide information concerning radioactivity levels in systems throughout the plant, to indicate radioactive leakage between systems, to monitor equipment performance, and to monitor and control radioactivity levels in plant discharges to the environment. Certain liquid and gaseous streams will be continuously monitored for radioactivity. Monitors on selected effluent release lines will automatically terminate discharges should radiation levels exceed predetermined values. Table 11.3.1 indicates the proposed locations, types of continuous monitors to be used, and the monitors which will provide for automatic termination of discharges. Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be sampled and analyzed in the plant laboratory.

We have reviewed the locations and types of effluent and process monitoring and sampling to be provided. Based on the plant design and on the continuous monitoring locations and continuous and intermittent sampling locations, we conclude that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which affect radioactivity releases. On this basis, we conclude that the monitoring and sampling provisions satisfy the requirements of General Design Criteria 13, 60 and 64 and the guidelines of Regulatory Guide 1.21 - Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, and are acceptable.

11.4 Evaluation Findings

Our review of the proposed liquid, gaseous and solid radwaste systems included system capabilities to process the types and volumes of wastes expected during normal operations including anticipated operational occurrences in accordance with General Design Criterion 60, the design provisions incorporated to control releases of radioactive material due to leakage overflows or hydrogen explosion in accordance with General Design Criterion 60, and the quality group and seismic design classification in conformance with the guidelines of the Effluent Treatment Systems Branch Technical Position ETSB 11-1. We have reviewed the applicants' system descriptions, process flow diagrams, piping and instrumentation diagrams, and design criteria for the components of the radwaste treatment systems and for those auxiliary supporting systems that are essential to the operation of the radwaste treatment systems. We have performed an independent calculation of the expected releases of radioactive materials in liquid and gaseous effluents based on the calculational methods of Draft Regulatory Guide 1.CC. 307188

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TABLE 11.3.1

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Stream Monitored

Liquid*

Component Cooling Water Loops
Service Water Discharge
Liquid Radwaste Discharge+
Cooling Tower Blowdown Discharge

Gas**

Offgas Discharge+
Containment and Drywell Ventilation Exhaust+
Radwaste Building Vent
Auxiliary Building Vent+
Fuel Building Vent+
Turbine Building Vent
Plant Exhaust Duct
Clean Steam (auxiliary steam reboilers)

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*All liquid streams will be monitored for gross gamma activity

**All gas streams will be monitored for noble gas (beta or gamma); other forms of radioactivity are sampled for laboratory analysis.

+These monitors provide annunciation and automatic closure of isolation valves terminating releases or diversion to alternate systems when the radiation level exceeds a predetermined valve.

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Our review of the process and effluent radiological monitoring and sampling systems included the provisions proposed for: (1) sampling and monitoring all plant effluents in accordance with General Design Criterion 64; (2) providing automatic termination of effluent releases and for assuring control over discharges in accordance with General Design Criterion 60 and Regulatory Guide 1.21; (3) sampling and monitoring plant waste process streams for process control in accordance with General Design Criterion 63; and (4) conducting sampling and analytical programs in accordance with the guidelines of Regulatory Guide 1.21. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

We conclude that the liquid and gaseous waste treatment systems and the ventilation systems will reduce radioactive effluents to as low reasonably achievable levels in accordance with Section 50.34a 10 CFR 50, Appendix I to 10 CFR Part 50, and the Annex to Appendix I to 10 CFR Part 50.

Based on the above described evaluation, we conclude that the proposed radwaste treatment and process and effluent radiological monitoring systems are acceptable. The basis for acceptance has been conformance of the applicants' designs, design criteria, and design bases for the radioactive waste treatment and monitoring system to the applicable regulations and guides referenced above, as well as to staff technical positions and industry standards.

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12.0 RADIATION PROTECTION

12.1 Shielding

The applicants' design objectives are to ensure that: (1) radiation exposure levels will be kept as low as reasonably achievable; (2) the criteria specified in 10 CFR Part 20 and other applicable regulations will be met during normal operation including anticipated operational occurrences; (3) the guidelines specified in 10 CFR Part 100 will be met during postulated accidents; and (4) exposures to plant personnel will be within the requirements of General Design Criterion 19, in the unlikely event of an accident. These objectives are consistent with our acceptance design objectives. In meeting these objectives, the applicants have had the benefit of their experience in operating the Millstone Nuclear Power Station, Unit 1, along with their architect engineer's experience in designing other nuclear power stations.

In response to our request, the applicants have: (1) stated a clear management policy to maintain occupational exposures as low as reasonably achievable; (2) identified the specific corporate structures related to responsibility for implementation of that policy; and (3) specified in detail those facility and equipment design considerations directed to assure its accomplishment.

The applicants are committed to ensuring that plant operations are conducted in such a manner as to maintain personnel occupational radiation exposures as low as reasonably achievable, in accordance with Regulatory Guide 8.8 - Information Relevant to Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable (Nuclear Power Reactor). This objective will be achieved through administrative exposure control procedures, adequate work planning, and safe practices in all activities related to unit operation. The general arrangements and shielding provisions are in accordance with Regulatory Guide 8.8, and are designed to provide levels of exposure to operating personnel that are as low as reasonably achievable.

Whenever possible, piping systems that will contain radioactive material will be routed so as to minimize exposure to plant personnel. This procedure will include: (1) separating high radiation level from low radiation level piping for maintenance purposes; (2) use of shielded pipes in low radiation level areas if routing through the area cannot be avoided; and (3) avoiding the routing of high radiation level pipes in corridors. High radiation level equipment will be located away from high usage personnel traffic ways in the containment, turbine building, auxiliary building, and radwaste building. Typical high usage passageways are the control rooms, the outside doorways, the stairways, and the elevator. Provision will be made for shielding of major equipment during inservice inspection, maintenance, and instrumentation calibration. Motor operated or diaphragm valves will be used on high level radiation piping where practical to assure low exposure to personnel at control stations. To prevent radiation streaming, penetrations will be oriented so as not to pass through a shield wall in a direct line with equipment or other piping with a high radiation source. Effort will be made to locate processing equipment and systems in such a manner as to minimize the need for shielding. Labyrinths will be used to eliminate streaming from equipment. To the extent posible, instruments and valves requiring surveillance and maintenance will be located outside of high radiation areas.

All plant areas will be classified into radiation zones. There will be six such zone classifications, all based on limiting personnel occupation time in radiation areas, thereby maintaining occupational radiation exposures as far below the 10 CFR Part 20 limits as reasonably achievable.

The applicants' calculations of source terms are based on the General Electric data on operating plants over the past decade. The radionuclides included are those considered significant to one or more of the following criteria:

(1) Plant equipment design.

(2) Shielding design.

(3) Understanding system operation and performance.

(4) Measurement practicability.

(5) Evaluating radioactive material releases to the environment.

The design basis noble radiogas source term was established at an annual average rate of 0.1 curies per second (t=30 minute), using 1970 and 1971 KRB and Dresden 2 data to model the nuclide mixture. For halogens, the design basis source term was based on 700 microcuries per second of iodine-131. Design basis concentrations in the coolant were conservatively estimated from experience. The applicants have provided the location, size and shape of significant sources of radiation in the auxiliary building, radwaste building, fuel building, turbine building, and containment structure.

The basic radiation transport analysis method used for the applicants' shield design is based on the Discrete Ordinates and Point Kernel methods. The design approach is similar to that described in the topical report, "Stone and Webster Radiation Shielding Design and Analysis Approach," which has been reviewed and found acceptable by the NRC staff. The applicants used information gained from operating stations to improve mathematical and physical models. The assumptions used in their shielding calculations are conservative and are acceptable to the staff.

The applicants' area radiation monitoring system is designed to: (1) monitor the radiation levels in areas where personnel may be required to work; (2) alarm when the radiation levels exceed preset levels to warn of abnormal radiation levels in areas where radioactive material may be present; and (3) provide a continuous record of radiation levels at key locations throughout the plant. In order to meet these objectives, the applicants plan to use thirty area monitors located in areas where personnel may be expected to remain for extended periods of time. The above objectives and location criteria are in conformance with 10 CFR Part 50 and Part 70. The area radiation monitoring system is equipped with two checking systems to test the integrity of the system, local and remote audio and visual alarms, and a facility for central recording.

As low as reasonably achievable design concepts will be used throughout the design and construction of the plant. Penetrations in shield walls will be made as high on the wall as practicable to minimize personnel exposure. All radioactive field run process piping will be positioned to limit exposure to plant personnel. Where necessary and practicable, interior surfaces of piping and ductwork will be designed to minimize contamination buildup. Changes to layouts will be reviewed with regard to radiation exposure, maintenance, operability, and access.

The applicants have based their estimate of annual man-rem exposure on their experience from currently operating boiling water reactors, their own operating experience, and improvements in design of systems to maintain in-plant radiation levels as low as reasonably achievable. Their estimate of 430 man-rems per year for the two Montaque units includes some maintenance personnel exposures, but does not include contractor personnel exposures.

This exposure level is associated with normal operational exposures such as from routine plant maintenance, routine plant operation, inservice inspection, radioactive waste handling, technical services functions, and from refueling, maintenance and operations. Our studies of currently operating light water reactors show that they average 400-500 man-rem per unit annually, with 80 percent of this exposure being due to major maintenance activities. The applicants' exposure estimates are consistent with our figures and the staff's as low as reasonably achievable policy. We, therefore, find them acceptable.

12.2 Ventilation

The applicants' proposed ventilation systems will be designed to provide ventilation air for the station's buildings and to ensure that airborne concentrations to which personnel may be exposed are well below those specified in 10 CFR Part 20. The applicants intend to meet these objectives and maintain personnel exposure as low as reasonably achievable by: (1) maintaining air flow from areas with little or no potential radioactive contamination to areas with progressively greater potential radioactive contamination; (2) maintaining certain areas at slight, subatmospheric pressures to avoid the release and/or spread of airborne radioactive material to other areas; and (3) maintaining as low as reasonably achievable airborne concentrations by using sufficient airflow rates in all areas. These design criteria are in accordance with those given in Regulatory Guide 8.8 and are, therefore, acceptable.

The containment ventilation system will be capable of reducing the halogen concentration to approximately one maximum permissible concentration whenever containment entry is required. The air filtration system in the control room will be designed to limit radiation exposure to control room personnel in accordance with General Design Criterion 19. The ventilation system features mentioned above, and others presented in Section 9.4 of the Preliminary Safety Analysis Report, meet the applicants' design objectives and are acceptable.

The maximum expected radioactive airborne concentrations inside major plant buildings due to equipment leakage were considered in our review. The bases for these leakage calculations are given in Section 11.3.6 of the Preliminary Safety Analysis Report and are in accordance with Regulatory Guide 1.42-Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Plants. The airborne radioactivity monitoring system samplers will be located wherever airborne radioactivity can exist, or wherever routine access by operating personnel is expected.

In Section 12.2.6 of the Preliminary Safety Analysis Report, the applicants provide estimates of the inhalation and submersion dose rates to plant personnel in major plant buildings. These dose rates are derived from the applicants' airborne radioactivity source terms. Using reasonably conservative occupancy times, the applicants provide estimates of peak concentrations and annual doses from in-plant airborne radioactivity. We find these results to be acceptable.

12.3 Health Physics Program

The applicants' stated policy is to maintain personnel radiation exposures within the provisions of 10 CFR Part 20, and other applicable regulations, and beyond that, to keep them as low as reasonably achievable in accordance with Regulatory Guide 8.8.

The applicants' Technical Supervisor will be responsible to the Assistant Plant Superintendent. He will have overall responsibility for reactor engineering, computer engineering, chemistry, instrumentation and controls, and radiation protection. Directly under him will be the Radiation Protection Supervisor who will be responsible, with the assistance of five health physics technicians, for all radiation protection aspects of the plant. It will be the responsibility of this group to evaluate radiological conditions of system operations, establish procedures to be followed for the protection of all plant personnel, ensure that all applicable Federal and State regulations are met and that the required radiation protection records are adequately maintained. The proposed implementation of this program is in accordance with Regulatory Guide 8.8 and is acceptable.

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Some of the methods to be used in the radiation protection program to minimize personnel exposures include minimizing time spent in radiation areas through work planning and administrative controls, periodic scheduled radiation and contamination surveys in all plant areas, and periodic review and evaluation of radiation and contamination levels, personnel exposures, and the effectiveness of the radiation protection program.

The radiation protection facilities will include access control checkpoints, change and decontamination areas, health physics station, chemical laboratory and counting room. We find that these facilities are sufficient to maintain occupational radiation exposures as low as reasonably achievable.

For radiation protection purposes the applicants propose to use protective clothing, respiratory equipment, air sampling equipment, portable radiation measuring instruments, calibration sources, counting room instrumentation, area monitors, airborne activity monitors, laboratory equipment, and special shielding materials. We find that the applicants will be able to maintain occupational radiation exposure as low as reasonably achievable using this equipment.

All plant personnel will be required to wear a film badge. Those individuals not restricted to the office area will also be issued a direct reading dosimeter. Neutron film badges, neutron dosimeters, and alarming dosimeters will also be provided for personnel when necessary. All radiation exposure information will be processed and recorded in accordance with the requirements of 10 CFR Part 20.

Based on the information presented in the application including the applicants' responses to our requests for additional information, we conclude that the applicants intend to implement a radiation protection program that will maintain in-plant exposures within the applicable limits and will keep occupational radiation exposures as low as reasonably achievable.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicants

The Northeast Nuclear Energy Company, acting as agent for the owners, is responsible for the design, construction, and operation of the Montague Nuclear Power Station. They are a subsidiary of Northeast Utilities. General Electric Company will be responsible for the nuclear steam supply systems and Stone and Webster Engineering Corporation will provide engineering and construction services.

Northeast Utilities Services Company, another subsidiary of Northeast Utilities, will implement Northeast Nuclear Energy Company's responsibilities for the design and construction of the Montague Nuclear Power Station. Within Northeast Utilities Services Company, the overall direction of the project will be implemented by the Manager, Nuclear Project, who is responsible to the Vice-President, Engineering and Construction. Quality Assurance aspects of the project are discussed in Section 17.0 of this report.

The proposed organization for facility operation will consist of a technical staff of approximately 163 persons for two unit operation, under the direction of a Plant Superintendent and an Assistant Plant Superintendent. Reporting to the Assistant Plant Superintendent will be: an Operations Supervisor responsible for the day-to-day activities of the operating shifts with a staff of approximately 52 persons; a Maintenance Supervisor responsible for all mechanical and electrical maintenance within the plant with a staff of approximately 43 persons; a Technical Supervisor responsible for providing technical support, with a staff of approximately 48 persons; and a staff of approximately 10 technical assistants. Reporting directly to the Plant Superintendent will be a Training Coordinator, and a Quality Control Group headed by a Quality Control Supervisor. This is a conventional type of plant organization for plant operations. The shift crew for each unit will consist of five persons, one of whom will be a licensed senior operator and two of whom will be licensed operators.

The applicants have stated that the qualification requirements of station personnel will meet the requirements set forth in ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel." This meets the NRC staff recommendations stated in Regulatory Guide 1.8, "Personnel Selection and Training."

Technical support for the plant staff will be provided primarily by the System Production Department of Northeast Utilities Services Company.

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We conclude that the applicants have established an acceptable technical organization to implement their responsibilities for the design and construction of the Montague Nuclear Power Station and that the proposed plant organization, the proposed qualifications of personnel, and the proposed plans for offsite technical support are acceptable.

13.2 Training Programs

The Plant Superintendent will have overall responsibility for the conduct and administration of the station training program. A Training Coordinator will be responsible for coordinating all phases of training; however, responsibility for preparation of detailed training programs will rest with individual department heads.

The applicants state that the objectives of the training program are:

(1) To train a staff to operate and maintain the nuclear units safely, dependably, and economically.

(2) To prepare the technical service groups for their functions necessary to support and assure safe operation.

(3) To prepare shift supervisors and control room personnel for the NRC qualification examinations for reactor operator and senior reactor operator licenses.

Portions of the training program will be similar to the program used at Millstone, Unit 2. Instructors used in the training program will be qualified members of the in-house supervisory and technical staff, Northeast Utilities Services Company engineering staff, and from the General Electric Company and other vendor or consultant specialists as required.

It is anticipated that most of the participants in the licensed operator training program will possess a wide variety of previous nuclear training. These personnel will receive training consisting of the following discrete segments: Site School (formal onsite classroom lecture series); Practical Reactor Operation; Reactor Simulator Training; Test and Plant Procedure Preparation; and Component and System Tests/ On-the-Job Training.

Maintenance and technical staff personnel will receive training in certain specific skills. All station personnel will receive General Employee Training consisting of a continuing program covering plant organization, employee benefits, company policies, station security requirements and general plant familiarization. In addition, before an employee is assigned to full time duties, he will be given a radiological health and safety course. This training will be supplemented by radiation emergency exercises and fire drills, resuscitation and respiratory equipment retraining, and Radiation Protection Manual review.

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Complete records of all training administered will be maintained.

The information submitted by the applicants relative to the proposed training program is satisfactory at this construction permit stage of our review to provide reasonable assurance that qualified individuals will be available for the preoperation test program, for operator licensing and for fuel loading.

13.3 Emergency Planning

The applicants have described their preliminary plans for coping with emergencies. In the Final Safety Analysis Report they will identify and evaluate a spectrum of accidents which might occur during the operation of the nuclear plant and which might affect plant personnel and/or members of the public. Action levels will be selected to assure that protective measures are initiated when necessary. The Shift Supervisor will direct the implementation of the Emergency Plan in accordance with detailed emergency procedures. The normal operating crew will be qualified to perform those actions necessary to implement the Emergency Plan. Offsite personnel will augment the operating crew, as required. An Emergency Control Center where emergency team members report for assignment will be established. The applicants have identified the notification responsibilities within the organization to ensure prompt and effective communications in the event of an emergency.

Initial contacts and arrangements will be made with the following agencies: Massachusetts Department of Public Health; Massachusetts Civil Defense; Massachusetts State Police; local police and fire departments; a local ambulance service; the Energy Research and Development Administration Brookhaven Area Office; Franklin County Public Hospital; Farren Memorial Hospital; and Yankee Atomic Electric Company. The Massachusetts Department of Public Health has the primary responsibility for planning for radiological emergency response in the environs of the plant. The State Police will institute protective measures as required, including evacuation of the public from any affected area.

The applicants have performed preliminary analyses to confirm the practicability of taking protective measures, including evacuation, within and beyond the site boundary during the expected lifetime of the plant. This effort is incomplete at this time, however. The applicants are actively working with the State of Massachusetts in the development of detailed response procedures for the Montague site. These include methods for the notification of the population-at-risk, and for physical evacuation, if necessary, of persons in the environs around the Montague Nuclear Station.

Two first aid rooms will be equipped with the necessary items to provide emergency first aid to injured personnel. One room will be designed to handle contaminated individuals. Ambulance service will be available locally. In addition, two company vehicles can be used for transfer of injured personnel to a hospital. Preliminary arrangements have been initiated with two area hospitals to treat contaminated injury cases.

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All plant operators, technical and maintenance personnel will receive training in emergency procedures. Exercises are planned to familiarize each member of the plant staff with the Emergency Plan and in particular, each individual's immediate action and responsibilities during a radiation emergency. A training program covering certain aspects of radiological emergencies and site familiarity will be made available to those offsite agencies whose services may be required in emergency situations.

The plant will be designed and will incorporate features to assure the capability of plant evacuation and of reentry to mitigate the consequences of an accident, including radiation emergency alarms, communications systems, and evacuation routes. The plant control room will be designed for continuous occupancy during and following the most severe accidents as analyzed in Section 15 of the application.

As stated above, the analyses to confirm the practicability of taking protective measures in the environs of the plant are not yet complete. The staff defers making a finding as to the acceptability of the applicants' emergency planning program until this information is available for review. We will address this matter in a supplement to this report as indicated in Section 1.8 of this report.

13.4 Review and Audit

The applicants have described their preliminary plans for review and audit of plant operations in Section 16.6 of the application. We conclude that these plans satisfy the the provisions described in ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are acceptable for the construction permit stage of our review.

13.5 Plant Procedures and Plant Records

Operating, maintenance and testing activities will be conducted in accordance with approved, written procedures. All station procedures developed will meet, as a minimum, the provisions of Regulatory Guide 1.33 - Quality Assurance Program Requirements and ANSI N18.7. Procedures will be reviewed by the Plant Operations Review Committee and approved by the Plant Superintendent.

Procedures used to support plant operation will be targeted for completion a minimum of three months prior to fuel loading with Administrative Control Procedures targeted for completion a minimum of one year prior to fuel loading.

The information submitted relative to these subjects is acceptable for the construction permit stage of our review.

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The applicants have stated that plant records will be maintained in accordance with Section XVII of Appendix B to 10 CFR Part 50. We conclude that these record keeping provisions are acceptable.

13.6 Industrial Security

The applicants have provided a general description of plans for protecting the plant against potential acts of industrial sabotage. Provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment have been described. We find that they conform to Regulatory Guide 1.17 - Protection of Nuclear Power Plants Against Industrial Sabotage. We conclude that the applicants' arrangements for protection of the plant against acts of industrial sabotage are satisfactory for this construction permit stage of our licensing process.

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14.0 INITIAL TESTS AND OPERATION

The initial test program for Montague 1 and 2 will be conducted by the applicants. Personnel of the applicants will receive technical direction and support from the nuclear steam supply system vendor (General Electric Company) and architect-engineer-constructor (Stone and Webster Engineering Corporation). The applicants have committed to develop and execute the test program in accordance with Regulatory Guide 1.68, "Preoperational and Initial Test Programs for Water-Cooled Power Reactors." The applicants' plans for the program are acceptable and should provide for verification of the functional adequacy of the facility. The details of the test program will be reviewed by the staff during the operating license stage of our review.

We conclude that the applicants have made acceptable plans for the staffing, development and conduct of the initial test programs.

15.0 ACCIDENT ANALYSIS

15.1 General

Two basic groups of events pertinent to safety are separately evaluated in this section. They include the abnormal operational transients and the accidents. In order for the analysis of events in either group to be acceptable, it is required that an accurate model of the reactor core be used, and that all appropriate systems whose operation (or postulated misoperation) would affect the event be included. For the transients, the analytical results must show no fuel damage and no damage to the reactor coolant pressure boundary. For accidents, which are far less probable, analysis results are allowed to show some fuel damage. These are analyzed to determine the extent of fuel damage expected and to assure that reactor coolant pressure boundary damage, beyond that assumed initially by the accident, will not occur.

The acceptability criteria of the analytical results for the transients are that no damage occurs to the fuel cladding (a sufficient, but not necessary, condition to meet this requirement is that minimum critical power ratio remain above 1.07), and that peak reactor vessel pressure does not exceed 110 percent of the design pressure (ASME Codes, Section III, Class I are met if nuclear system pressure remains below 1375 pounds per square inch gage, which is 110 percent of the 1250 pounds per square inch gage design pressure). These two requirements demonstrate, respectively, that the first radioactive material barrier (the clad) and the second barrier (the pressure vessel) are protected during abnormal operational transients.

For design basis accident analyses, which evaluate situations that require functioning of the engineered safety features (including containment), it is necessary only to demonstrate that the second barrier (the pressure vessel) is protected. This is accomplished by ensuring that peak fuel enthalpy remains below 280 calories/gram. This limit conservatively assures the absence of any destructive pressure pulses due to fuel vaporization, thereby assuring minimal core damage. The cladding must remain essentially intact (even though some perforations are allowed) if a coolable geometry is to be maintained.

15.2 Abnormal Operational Transients

Abnormal operational transients are the result of single equipment failures or single operator errors that can reasonably be expected during any mode of operation. The applicants have provided analyses of various abnormal operational transients in the application. These analyses include such events as process system control malfunctions, inadvertent control rod withdrawal, turbine trip, loss of electrical load, and variations in operating parameters.

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Eight nuclear system parameter variations are listed as potential initiating causes of challenges to the integrity of the fuel cladding and to the reactor coolant pressure boundary. These parameter variations in the analyzed transients are as follows:

(1) Nuclear System Pressure Increase. Transients analyzed in this group included loss of load events such as generator trip, turbine trip, loss of condenser vacuum, closure of one or all of the main steam line isolation valves, and malfunction of the reactor primary system pressure regulator.

(2) Reactor Water Temperature Decrease. These transients included events that might cause a power surge by reduction of the reactor primary coolant water temperature. They include malfunctions of the feedwater control system in a direction to increase feedwater flow, loss of a feedwater heater, shutdown cooling malfunction, and inadvertent activation of an auxiliary cold water system.

(3) Reactivity Insertions. These transients included rod withdrawal transients from zero reactor power, hot critical condition, and from full power; fuel assembly insertion errors; and control rod removal errors during refueling.

(4) Reactor Water Inventory Decrease. These transients included events leading to a decrease in the inventory of reactor primary coolant such as loss of auxiliary power, loss of feedwater, pressure regulator failure in a direction to cause decreasing reactor system pressure, inadvertent opening of a safety/relief valve, and opening of condenser bypass valves.

(5) Primary Coolant Flow Decrease. These transients included failure of one or more recirculation pumps or malfunction of the recirculation flow control system in a direction to cause decreasing flow.

(6) Reactor Coolant Flow Increase. These transients included events that might increase the recirculation flow and thus induce a positive reactivity increment. They included a malfunction of the recirculation flow controller in a manner to cause increasing primary coolant flow and the start-up of a recirculation pump that had been on standby.

(7) Core Coolant Temperature Increase. The transient analyzed in this category was loss of shutdown cooling.

(8) Excess Coolant Inventory. The transient analyzed in this group was feedwater controller failure to maximum demand.

The applicants' transient analyses included effects due to the prompt relief trip system. The applicants have commited to the incorporation of all amendments to the GESSAR-238 Nuclear Island Standard Design application, applicable to the nuclear steam supply system, through the time of issuance of the Supplements to the Safety Evaluation Report for GESSAR reporting the resolution of outstanding items described in the December 1975 Safety Evaluation Report, NUREG-75/110. The GESSAR application has adopted a "fast scram" system in place of the prompt relief trip and the applicants' commitment to GESSAR thus commits them to "fast scram" as well. In the transmittal letter for Amendment 14 Supplement 7, the applicants have indicated that if the prompt relief trip system is reviewed and approved by the NRC for use in the BWR-6 reactors they would consider using this system for the Montague 1 and 2 reactors. The transient analyses discribed in Section 15 have been repeated with the "fast scram" system and are reported in the GESSAR application and are applicable to the Montague 1 and 2. We find it acceptable to defer our review of the modification and test program for the fast scram system which will be completed prior to a decision for issuance of the Montague 1 and 2 construction permits. This matter and its resolution will be provided in a supplement to this report.

15.2.1 A System for Increasing the Negative Reactivity Insertion Rate

The need for increasing the negative reactivity insertion rate is related to the operating objective of General Electric and the utility applicants to continue full rated power operation into the end-of-cycle-life period. From current experience with BWR operation it may be expected that Montague 1 and 2 and similar plants with control rod drive systems of current design may find it necessary to reduce power somewhat during the last 10-20 percent of each cycle. This situation will occur, if it occurs at all, as the reactor core approaches its equilibrium fuel cycle.

In the event of a sudden loss of normal heat removal capability which can occur as a result of loss-of-load transients such as turbine trip or generator trip, sudden reactor coolant system pressure increases will occur. The increased pressure causes collapse of steam voids that were present in the core, which in turn causes a power increase due to the positive reactivity effect of void collapse. This power increase then tends to further increase pressure. The above cycle of events is terminated by reactor scram (rod insertion), but toward end-of-fuel-cycle in boiling water reactors the time required to achieve effective power reduction from rod insertion is somewhat increased. This is because the rods have further to travel from their end-of-cycle position (mostly they are completely withdrawn from the core and ready for insertion from below the core) and because the rods must reach the more reactive region (which is nearer the top of the core at end-of-cycle) to achieve full effectiveness.

This operating condition has been studied by General Electric, the NRC staff and our consultant (Brookhaven National Laboratory). Analyses of core dynamics performed for certain events (such as turbine trip without bypass) for an equilibrium core operating at full power near its end-of-cycle, and assuming a number of conservative assumptions, show that without further analyses, boiling water reactors employing control rod drive systems of current design might require a limited decrease in acceptable power level during the last 10-20 percent of each near equilibrium operating cycle. The staff will complete its review of this matter on the GESSAR 238 Nuclear Island application (Docket No. STN 50-447) and on other applications in which the General Electric Company is the common supplier of the nuclear steam supply system. The applicant's have agreed to adopt the resolution arrived at on the GESSAR docket as indicated in Section 1.8 of this report. We find this acceptable.

15.2.2 Proposed Systems

If it is determined that additional negative reactivity insertion capability is needed or desired, design changes will be required if full rated power operation is desired during the end of each operating cycle as the core approaches its equilibrium reload pattern. Two design modifications have been proposed for increasing the negative reactivity insertion rate associated with the above transients. The two proposed modifications are:

(1) Prompt Relief Trip. This system would automatically open certain safety/relief valves immediately upon occurrence of a loss-of-load event. This would prevent or inhibit the cycle of pressure increase - void collapse - power increase - pressure increase by early relief of the initial pressure increase. This solution has been proposed for the Montague Nuclear Power Station, and earlier was proposed on the GESSAR-238 Nuclear Island application. Staff review under the latter application determined that there were several outstanding questions concerning the proposed system; e.g., the effect on peak clad temperature of steam venting through relief valves during a loss-of-coolant accident.

(2) Fast Scram. This solution is based on faster control rod insertion to achieve reactor shutdown before pressure or minimum critical power ratio limits are approached. This would be accomplished by increasing the size of the hydraulic piping in the control rod drive system and increasing the pressure in the scram accumulator. The design of the hydraulic control units will remain substantially unchanged. These design changes result in scram times on the order of 1.5 seconds (for 75 percent insertion) as compared with the present design of 2.78 seconds. General Electric has proposed confirmatory testing to demonstrate the system capabilities for the faster insertion and any effects it may have. We are currently reviewing the fast scram system on the GESSAR-238 application. We concluded that this matter is being resolved as a post-Preliminary Design Approval item on the GESSAR-238 application. The applicants have indicated in Amendment 14 they will adopt this system if the prompt relief trip described above is determined to be unacceptable.

15.2.3 Conclusions Regarding Prompt Relief Trip and Fast Scram Systems

We believe that one or both of these systems can be designed in a version that will be acceptable to the NRC staff and, if required for that purpose will provide for full power operation out to the end of the planned operating cycle with an equilibrium core even with the conservative analytical methods presently used. However, no safety

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requirements are dependent on the satisfactory design of either system. The NRC regulations require that the facility be operated within the safety limits and design conditions presented in the application. These are reviewed and approved by the NRC staff. They provide an adequate basis for protecting the health and safety of the public. The facility can be operated within these limits and conditions by adopting appropriate operating procedures during the end-of-cycle-life period (if necessary), in the same manner as those plants presently operating with control systems of current design. Further operational experience may determine a need for Montague 1 and 2 to operate at some reduced power level during the end-of-cycle period. On the other hand, the applicants may include in the station design additional equipment (e.g., prompt relief trip or fast scram) for increasing the negative reactivity insertion rate in order to achieve the operational objective of a higher power level for the extended periods desired by General Electric and the utilities. Inclusion of this additional equipment may be made to improve station operational characteristics and is not necessary to achieve any safety requirement. Therefore, we conclude that the decision to include one or the other of the proposed systems is not necessary for the construction permit stage of review. However, in view of the special schedules existing for this application we will complete our review of whichever of these systems is to be used prior to a decision for issuance of construction permits and will report on the selected design in a supplement to this report.

15.3 Design Basis Accidents

In order to demonstrate the effectiveness of the Montague engineered safety features, we computed offsite doses resulting from the postulated loss-of-coolant, fuel handling and control rod drop accidents. In addition, we have computed the doses resulting from the purging of the post-loss-of-coolant accident containment atmosphere. Our acceptance criteria are that the doses from these postulated accidents (as evaluated by the NRC staff) be within the exposure guidelines of 10 CFR Part 100. As recommended in Regulatory Guide 1.3, the doses considered appropriate at the construction permit stage should not be no more than 150 rem thyroid and 20 rem whole body. The offsite doses we calculated for these accidents are shown in Table 15.1, and the assumptions we used are listed in Table 15.2. All potential doses calculated by us at the exclusion boundary for the postulated accidents are within the 10 CFR Part 100 guideline values. In these analyses, the low penetrating beta radiation has been treated as a skin dose, rather than a contributor to the whole body dose.

The charcoal filters of the standby gas treatment system have been given credit for 99 percent efficiency in removal of all species of iodine during the loss-of-coolant, because they have a bed depth of at least four inches and comply with Regulatory Guide 1.52 - Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units for Light Water Cooled Nuclear Power Plants. We have evaluated the consequences of a postulated loss-of-coolant accident without giving credit for mixing within the annulus.

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TABLE 15.1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary (1250 meters)		Course of Accident Low Population Zone (4022 meters)	
Accident	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss-of-Coolant	156	16	119	3
Main Steam Isolation Valves Leakage			7	< 1
Total Loss-of-Coolant Accident Event			126	3
Post-Loss-of-Coolant Accident Hydrogen Purge Dose	Not applicable		19	1
Fuel Handling	8	4	< 1	< 1
Control Rod Drop Accident*	< 1	< 1	< 1	< 1

* We are in the process of revising our evaluation of the rod drop accident, and expect that the potential consequences will be reduced, because of more realistic assumptions. This will not affect our conclusions that potential radiological consequences of this accident are acceptable.

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TABLE 15.2

ASSUMPTIONS USED IN THE ESTIMATE OF DESIGN BASIS ACCIDENT DOSES

LOSS-OF-COOLANT ACCIDENT

Power Level	3730 megawatts thermal
Operating Time	3 years
Primary Containment Leak Rate*	0.275 percent
Iodine Composition	
Elemental	91 percent
Particulate	5 percent
Organic	4 percent
Filter Efficiencies	
Elemental	99 percent
Particulate	99 percent
Organic	99 percent

X/Q VALUES, seconds per cubic meter

0-2 hours @ 1250 meters	8.4×10^{-4}
0-8 hours @ 4022 meters	5.6×10^{-5}
8-24 hours @ 4022 meters	3.8×10^{-5}
24-96 hours @ 4022 meters	1.7×10^{-5}
96-720 hours @ 4022 meters	5.1×10^{-6}

* 2-1/2 percent of the primary containment leak rate is assumed to bypass the filters and exit directly to the atmosphere.

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TABLE 15.2 (Cont'd.)

POST-LOCA HYDROGEN PURGE DOSE

Power Level	3730 megawatts thermal
Volume of Primary Containment	1.64×10^6 cubic feet
Purge Duration	30 days
Holdup Time in Containment Prior to Purge Initiation	19 days
Filter Efficiencies	
Elemental	99 percent
Organic	99 percent
Purge Rate	34 standard cubic feet per minute

X/Q VALUES, seconds per cubic meter

96-720 hours @ 4022 meters	5.1×10^{-6}

FUEL HANDLING ACCIDENT

Power Level	3730 megawatts thermal
Shutdown Time	24 hours
Total Number of Fuel Rods in the Core	46,116
Number of Fuel Rods Involved in the Refueling Accident	98
Iodine Fractions Released from Pool	
Elemental	75 percent
Organic	25 percent
Filter Efficiencies	
Elemental	95 percent
Organic	95 percent

X/Q VALUES, seconds per cubic meter

0-2 hours @ 1250 meters	8.4×10^{-4}
0-8 hours @ 4022 meters	5.6×10^{-5}

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TABLE 15.2 (Cont'd.)

CONTROL ROD DROP ACCIDENT

Power Level	3730 megawatts thermal
Number of Fuel Rods Involved	770
Fraction of Fission Product Inventory	
Released to Coolant	
Noble Gases	100 percent
Iodines	50 percent
Iodine Fraction Released	
to Condenser	10 percent
Iodine Partition Factor and Plate Out	
in the Turbine and Condenser	10 percent
Condenser Leak Rate	0.5 percent per day

X/Q VALUES, seconds per cubic meter

0-2 hours @ 1250 meters	8.4×10^{-4}
0-8 hours @ 4022 meters	5.6×10^{-5}
8-24 hours @ 4022 meters	3.8×10^{-5}

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On the basis of our experience with the evaluations of the steam line accident for plants of similar design, we have concluded that the consequences of this accident can be controlled by limiting the permissible radioactivity concentrations in the reactor coolant so that potential offsite doses are small. We will include limits in the technical specifications on the coolant activity concentrations such that the potential two-hour doses at the exclusion radius, as calculated by the NRC staff for these accidents, will be appropriately small fractions of the guideline values of 10 CFR Part 100. In keeping with recently issued design guidance for radioactive waste management systems installed in lightwater-cooled nuclear power reactor plants, we will require that the charcoal delay beds be designed to the requirements of Effluent Treatment Systems Branch Position 11-1.

15.3.1 Loss-of-Coolant Accident

The design basis accident loss-of-coolant accident is the same as that used in the analysis of other boiling water reactors, in that a double-ended break in the largest pipe in the reactor coolant system, the recirculation line, is assumed. Our analysis is consistent with the conservative assumptions presented in Regulatory Guide 1.3. It should be noted that with the 0.275 percent per day leak rate proposed by the applicants, the total (direct containment leakage plus main steam line isolation valve leakage plus hydrogen purge dose) 0-30 day thyroid dose at the low population zone is consistent with the 150 rem thyroid guideline doses listed in Standard Review Plan Section 15.6.5 and Regulatory Guide Number 1.3 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors.

The assumptions we used in this analysis are given in Table 15.2. The resultant calculated loss-of-coolant dose levels are given in Table 15.1, and are judged to be within acceptable limits.

15.3.2 Fuel Handling Accident

In this accident, we assumed that a fuel assembly is dropped during refueling operations, and that as a result of the fall, 98 fuel rods are damaged. Our assumptions conform to Regulatory Guide 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident and Storage Facility for Boiling and Pressurized Water Reactors. Activity released to the environs is assumed to be released through the standby gas treatment system filters within a two-hour period. Other assumptions used in this analysis are given in Table 15.2. The resultant calculated two-hour doses are given in Table 15.1. We find these calculated doses to be within acceptable limits.

15.3.3 Control Rod Drop

We have evaluated the consequences of a control rod drop accident. The assumptions we used in this analysis are shown in Table 15.2. The resulting doses are shown in

Table 15.1 and are well within the guideline values given in 10 CFR Part 100. We are in the process of revising our evaluation of the radiological consequences of a control rod drop accident in order to incorporate more realistic assumptions. We expect the potential consequences will be reduced, therefore this will not affect our conclusion that the consequences will be within acceptable limits.

15.3.4 Hydrogen Purge Dose

Based on the data included in Table 15.2, the hydrogen purge dose is computed to be small compared to the 10 CFR Part 100 guidelines at the low population zone for the course of the accident. Assumptions used in this analysis are included in Table 15.2. We find these calculated doses will be within acceptable limits.

15.4 Postulated Radioactivity Liquid Release Due to Tank Failures

The consequences of component failures which could result in contaminated liquid releases to the environs were evaluated for components containing liquid radioactive materials located outside reactor containment. The scope of the review included the calculation of radionuclide inventories in station components at design basis fission product levels, the mitigating effects of the plant design, and the effect of site geology and hydrology.

The tank that is estimated to contain the highest quantity of activity is the regenerant neutralizer tank. This tank will have a volume of 25,000 gallons and is assumed to be 80 percent full with a liquid activity concentration of approximately one microcurie per milliliter (based on an offgas release rate of 350,000 microcurie per second after 30 minutes delay). The critical direction of groundwater flow will be toward a downgradient well 1.6 miles southwest of the plant. We evaluated the liquid transit time for radwaste leakage to the nearest downgradient well to be approximately 20 years. We estimate a ground water dilution factor of 29,200.

Considering radioactive decay over the 20 year transit time, a rupture of the regenerant neutralizer tank will give a concentration of less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2 for unrestricted areas.

Based on the foregoing evaluation, we conclude that the postulated failure of the tank with the highest level of radioactivity should not result in radionuclide concentrations in excess of the 10 CFR Part 20 limits at the nearest potable water supply. Therefore, it is not necessary for the applicants to incorporate additional provisions in their design to mitigate the effects of component failure involving contaminated liquids.

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16.0 TECHNICAL SPECIFICATIONS

The technical specifications of an operating license will define certain features, characteristics and conditions governing operation of a facility that cannot be changed without prior approval of the Nuclear Regulatory Commission. The technical specifications will be developed and evaluated at the operating license stage of our review. However, in accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions or other items which are determined as a result of the preliminary safety analysis and its evaluation to be probable subjects of the technical specifications for the facility, with special attention given for those items which may significantly influence the final design.

We have reviewed the proposed preliminary technical specifications presented in Section 16 of the application with the objective of identifying those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to those being developed or in use for plants of a design similar to Montague 1 and 2. We have not identified any items which require special attention at this stage of our review.

On this basis we conclude that the proposed preliminary technical specifications are acceptable.

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17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for Montague 1 and 2, including the quality assurance programs of the principal contractors, the Stone & Webster Engineering Corporation, the engineer-constructor, and the General Electric Company, the supplier of the nuclear steam supply systems, is contained in Section 17 of the application and in the applicants' responses to NRC staff requests for additional information. The quality assurance program has been revised in Supplements 1, 2, 3, 5, 6 and by the applicants' letters to the NRC dated August 1, September 19, and October 14, 1975.

Our evaluation of the proposed program is based on our review of these documents to determine that the quality assurance programs of the applicants' affiliate, the Northeast Utilities Service Company, and of the principal contractors comply with the requirements of Appendix B to 10 CFR Part 50.

The applicants rely on the services of Northeast Utilities Service Company during the design, procurement, and construction phases of the Montague project. Similarly, the applicants provide the services required during preoperational testing and during the operation phase. During the operation phase, the applicants will utilize the technical support services of Northeast Utilities Service Company. Northeast Utilities Service Company and the Northeast Nuclear Energy Company are wholly owned subsidiaries of Northeast Utilities, and they are responsible for implementing the quality assurance program for Montague. General Electric has established a program for the nuclear steam supply systems, and Stone & Webster has established a program for the balance of the plant. Northeast Utilities Service Company is responsible for the Montague quality assurance program through the construction phase and is organized to control and verify the programs of these principal contractors.

17.2 Northeast Utilities Service Company Organization

The organizational structure of personnel having quality related functions for Montague is shown in Figure 17.2. The Manager of Quality Assurance reports to the Assistant Vice President of Generation, Engineering, and Construction who in turn reports through his Vice President to the Executive Vice President of Engineering and Operations. This Executive Vice President is responsible for eng   ering, construc-ion, and the related quality assurance activities for the nuclear power plant. The Manager of Quality Assurance is responsible for developing, establishing, and verifying implementation of the Montague quality assurance program.

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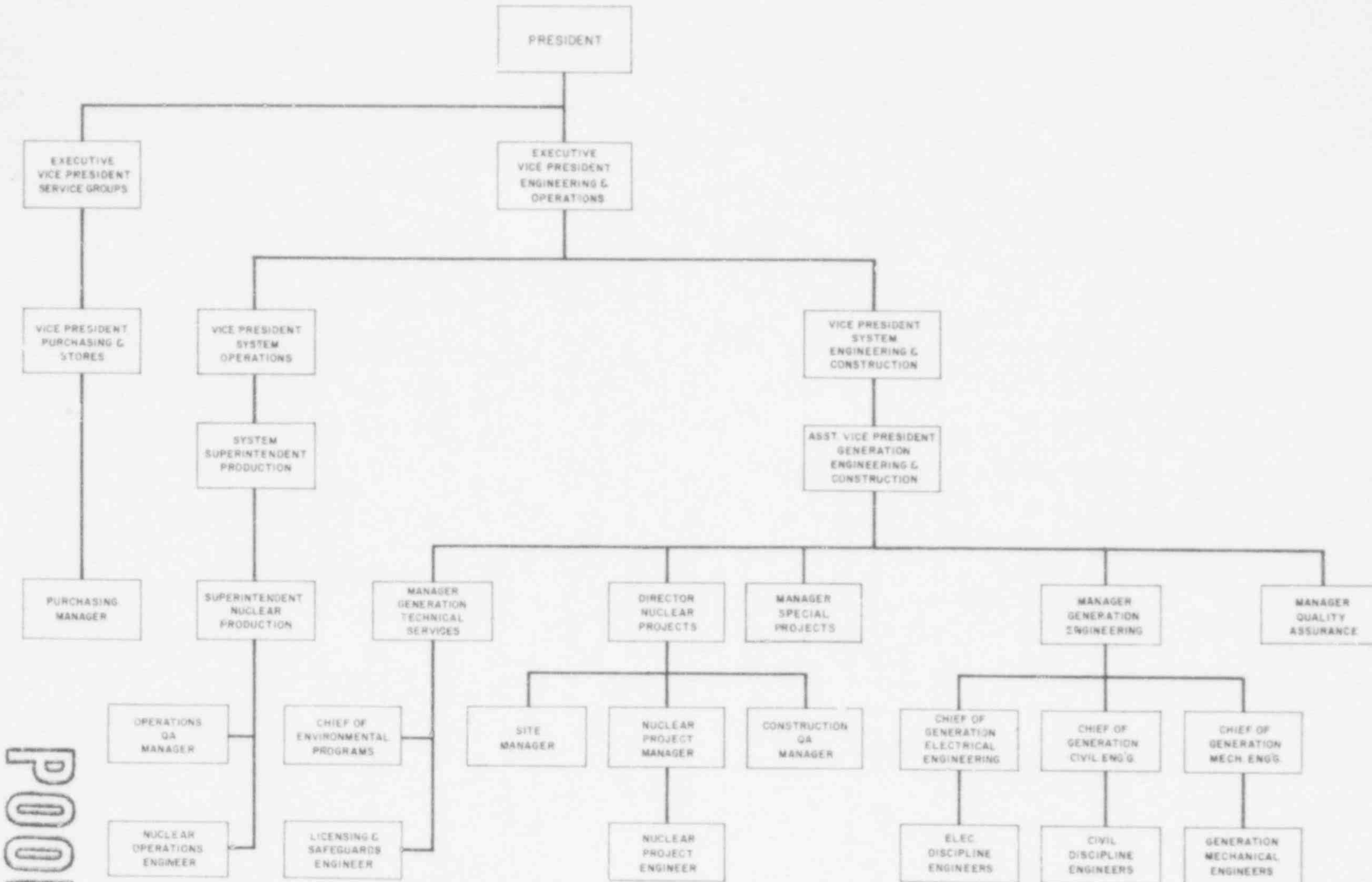

NUSCO QUALITY ASSURANCE ORGANIZATION
FIGURE 17.2

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Also shown in Figure 17.2 is the Construction Quality Assurance Manager, who does not report to the Manager of Quality Assurance. He reports to the Director of Nuclear Projects and is responsible for verifying proper performance of quality related activities at the Montague site during the construction phase. He also is responsible for documenting procedures and instructions for the construction quality assurance functions. These implementing procedures and instructions are found in the Northeast Utilities' quality assurance manual and Northeast Utilities Service Company's department procedures.

The Manager of Quality Assurance and the Construction Quality Assurance Manager are independent of those organizations whose quality related activities they verify. The Manager of Quality Assurance reviews construction quality assurance procedures and instructions and audits the construction quality assurance activities to maintain control over the entire quality assurance program. We find, based on the organizational structure shown in igure 17.2 that the organizations have adequate independence and report at a sufficiently high management level to accomplish their objectives.

The organizations implement their functions through their staff of engineers, specialists, and technicians. The Manager of Quality Assurance and the Construction Quality Assurance Manager and their staffs have authority to stop unsatisfactory work and to obtain resolution of quality problems.

We find that the review and approval of the construction procedures and instructions by the Manager of Quality Assurance provides adequate control of these procedures and instructions. In addition, we find that management, above the level of the quality assurance manager, has provisions to independently and regularly assess the scope, implementation, and effectiveness of the program to assure that it is meaningful and effectively complies with Appendix B to 10 CFR Part 50.

Since Northeast Utilities Service Company's description for implementing their program includes quality assurance stop work authority, quality assurance audit and followup responsibilities, we conclude that it is acceptable.

Based on our review of their quality assurance organization, we find that: (1) they are independent of the organizations whose work they verify; (2) they have clearly defined authorities and responsibilities; (3) they are organized so they can identify quality problems in organizations performing quality related work; (4) they can initiate, recommend, or provide solutions to quality problems; and (5) they can verify implementation of solutions. We, therefore, conclude that the Northeast Utilities Service Company organization complies with Appendix B to 10 CFR Part 50 and is acceptable.

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In Supplement No. 6 to the Preliminary Safety Analysis Report, the applicant provided a list of the Montague quality assurance program procedures along with a brief description of the purpose of each. A cross index to the related criteria of

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Appendix B to 10 CFR Part 50 is also given. Based on our review, we conclude that each criterion of Appendix B has been specifically included in written procedures within the quality assurance program.

The quality assurance manual is the responsibility of the Manager of Quality Assurance. It included a directive, signed by the Executive Vice President of Engineering and Operations, which requires compliance with the manual. Revisions to the manual are approved by the Assistant Vice President of Generation Engineering and Construction and the Manager of Quality Assurance.

The Preliminary Safety Analysis Report identified the structures, systems, and components that are subject to the program described in the quality assurance manual.

The applicants have committed to comply with the requirements of the Gray Book[1] and the Green Book.[2] These documents provide NRC guidance on quality assurance requirements during the design, procurement, and construction phases of nuclear power plants. Based on this commitment and the quality assurance program described in the application as amended, we find that the applicants' quality assurance program for design and construction is acceptable.

The Manager of Quality Assurance is responsible for assuring that indoctrination and training programs are established for personnel who participate in the quality assurance program. These programs are to assure that personnel performing quality affecting activities understand the purpose, scope, and method of implementing the program.

A comprehensive system of planned and documented audits is described in the application. The audit program includes the following types of audits to provide an independent verification and evaluation of the quality related procedures and activities to assure they comply with the applicants' quality assurance program:

(1) Audits of major contractors, including Stone and Webster and General Electric;

(2) Audits at the Montague plant site; and

(3) Audits internal to Northeast Utilities Service Company.

Audits are performed in accordance with written procedures or checklists by appropriately trained personnel having no direct responsibilities in the area audited. Audit results are documented and reported to appropriate levels of management for

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[1] "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants-Rev. 1," May 24, 1974 (WASH-1283 Rev. 1)

[2] "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH-1309)

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corrective action. Responses to audit findings are verified for implementation and effectiveness during followup audits. The staff finds Northeast Utilities Service Company's description of their audit activities to be acceptable.

17.3 General Electric Company

General Electric is responsible for providing the nuclear steam supply systems for Montague 1 and 2. Figure 17.3 shows the organization of the Boiling Water Reactor Operations of General Electric which provides nuclear plant services and equipment. This department operates under a Deputy Division General Manager who reports to the Vice President and General Manager of the Nuclear Energy Division. Reporting to this Deputy Division General Manager are Department General Managers and the Manager of Product and Quality Assurance Operation.

Each department and the Product and Quality Assurance Operation contain an organization specifically responsible for quality assurance which reports at a management level to assure independence consistent with Criterion 1 of Appendix B to 10 CFR Part 50.

Quality assurance management in each department and the Product and Quality Assurance Operation is free of prime responsibility for schedule or cost, has the authority to stop work pending resolution of quality matters, and has the freedom to: (1) identify quality problems; (2) initiate, recommend, or provide solutions to quality problems; (3) verify implementation of solutions; and (4) prevent further processing, shipment, installation, or utilization of nonconforming items until proper dispositioning has occurred. We find that General Electric has adequately defined the responsibilities of the organizations performing quality assurance activities and that they are acceptable.

The Deputy Division General Manager of the Boiling Water Reactor Operations has established a Quality Council which includes the managers of the major quality assurance organizations in the division. The Manager of Product and Quality Assurance Operation is Chairman of the Quality Council. The Quality Council, which meets quarterly, permits development of solutions to common quality related problems and provides a separate line of communication to top Boiling Water Reactor Operations management. In addition, the Manager of Product and Quality Assurance Operation audits the engineering, manufacturing, procurement, and construction organizations to assess the scope, implementation, and effectiveness of the quality assurance program.

The program applies to the safety related systems and components within the General Electric scope of work. General Electric has also committed to comply with the guidance provided by the NRC in the Gray Book and Green Book.

Though the basic scope of the quality system used by the various Boiling Water Reactor Operations organizations is the same, each functional organization has its own system of guides, procedures, instructions, and manuals that prescribes the methods for accomplishing its portion of the quality assurance program. Division instructions, prepared by the Product and Quality Assurance Operation and issued by authority of the

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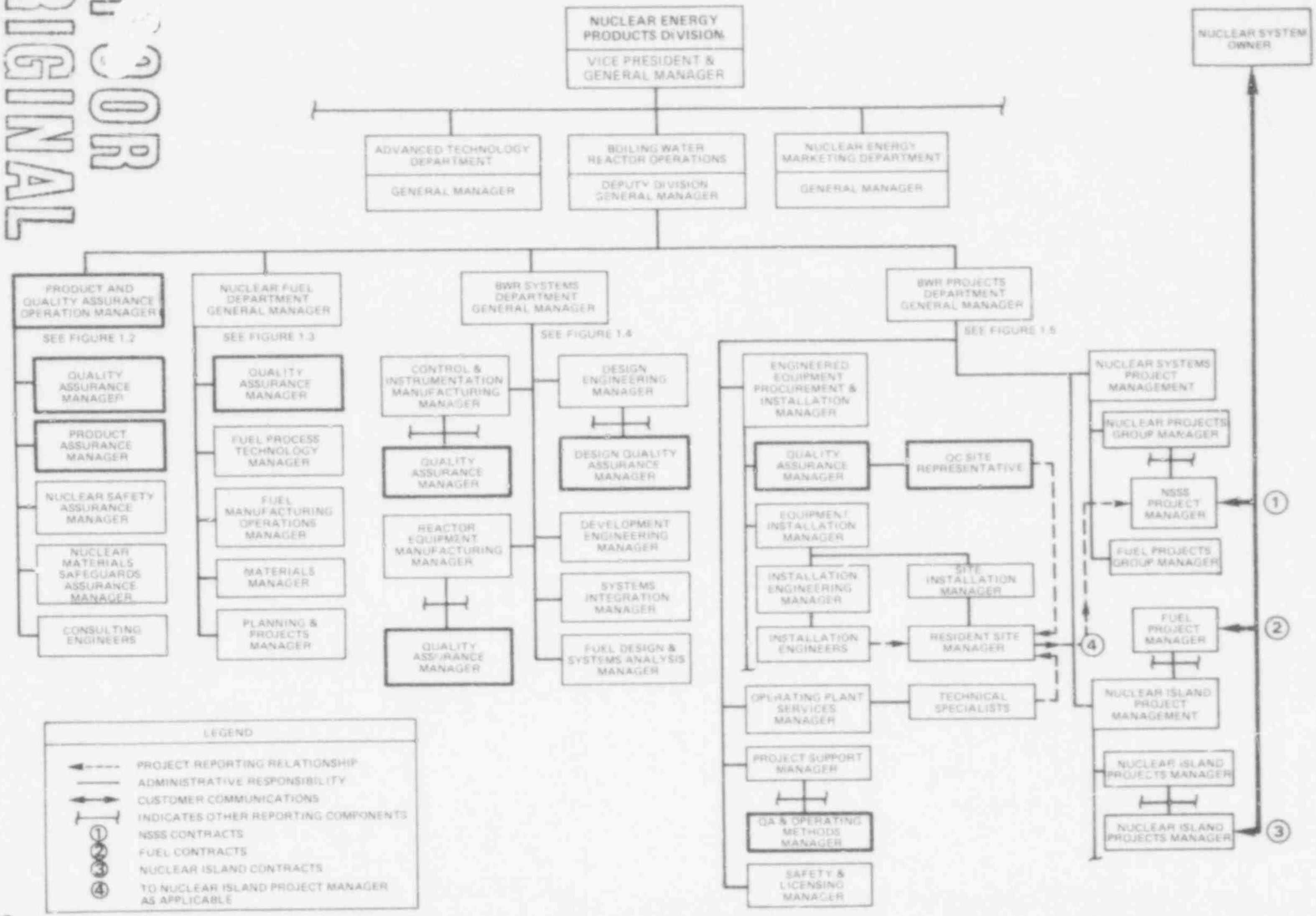

GENERAL ELECTRIC BWRO QUALITY ASSURANCE ORGANIZATION

Figure 17.3

General Manager of the Nuclear Energy Division, establish procedures and practices where a standardized, uniform approach is necessary for control.

A matrix which relates the procedures of the various manuals to the applicable quality assurance criteria of Appendix B to 10 CFR 50 is given in the application. Based on our review of this matrix, we conclude that each criterion has been specifically included in written procedures within the program.

The program included provisions for the control of design information. Design inputs are reviewed, and analyses are accomplished in accordance with applicable codes, standards, and regulatory requirements. Knowledgeable groups within General Electric including quality assurance personnel, independently review drawings and equipment specifications prior to issuance.

To provide for control over purchased items and services, General Electric evaluates the quality system of each prospective supplier of safety related items. Quality engineers review purchase requisitions, purchase orders, and subsequent change notices. General Electric reviews and retains supplier documentation which demonstrates acceptable quality. Audits and feedback of discrepancy data are used by quality engineers to measure supplier performance.

General Electric executes a comprehensive audit program which provides Boiling Water Reactor Operations management with information on the effectiveness of the program. General Electric audits activities affecting quality at General Electric and at supplier facilities. Audit areas include all quality related procedures and operations. Trained personnel not having direct responsibilities in the area being audited conduct the audits in accordance with defined procedures and checklists.

In our review, we have evaluated the General Electric quality assurance program for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the program includes an acceptable organization and contains the necessary provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and is acceptable for the nuclear steam supply systems for Montague 1 and 2.

17.4 Stone and Webster Engineering Corporation

Stone & Webster is the engineer-constructor of Montague 1 and 2. The Stone and Webster organization responsible for design, procurement, and construction activities is shown in Figure 17.4. The President has established quality assurance policies for the corporation. He has delegated the authority for development of the quality assurance program to the Vice President of Quality Assurance who is responsible for the overall program. As Figure 17.4, shows, the Vice President of Quality Assurance is independent of and has organizational authority equal to the other Stone and Webster Vice Presidents.

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STONE & WEBSTER ENGINEERING CORPORATION
QUALITY ASSURANCE ORGANIZATION

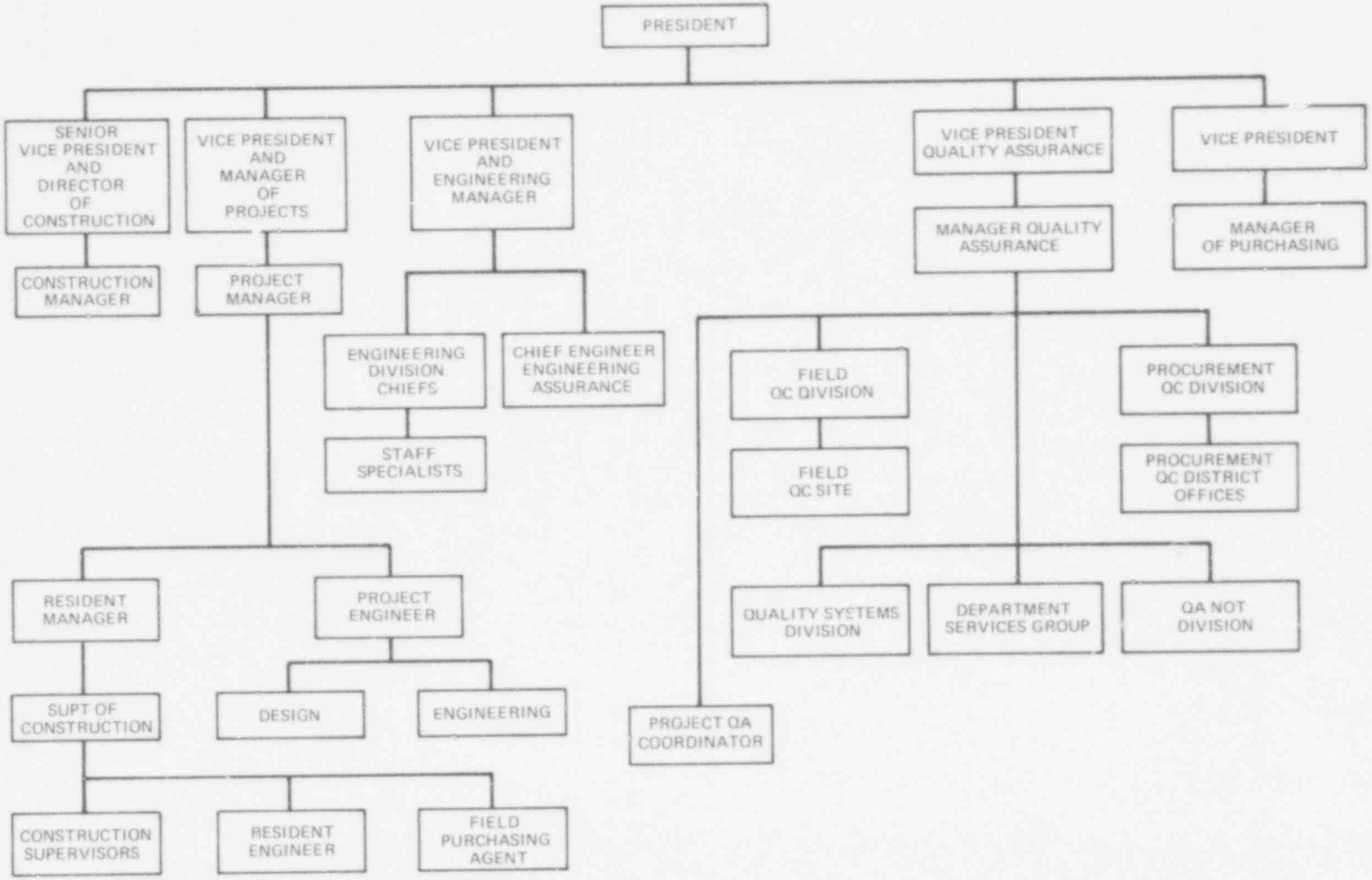

Figure 17.4

The Manager of Quality Assurance, who reports to the Vice President of Quality Assurance, is responsible for administering and managing the quality assurance program for procurement and construction activities. The Chief Engineer, Engineering Assurance Division of the Engineering Department, is responsible for administering and managing the quality assurance program for engineering and design.

Major activities which are carried out by the Quality Assurance and Engineering Assurance organizations are:

(1) The review and approval of design, procurement, manufacturing, inspection, construction, and test documents.

(2) Inspections and audits within the company, at supplier's facilities, and at the construction site.

Quality Assurance and Engineering Assurance personnel have the authority and freedom to identify quality problems; to initiate, recommend, or provide solutions; and to control further processing, delivery, or installation of a nonconforming item until proper disposition of the deficiency or unsatisfactory condition has been approved. We conclude that the organization is structured such that an individual is responsible for coordinating the direction and control of the quality assurance and quality control function and that personnel performing quality assurance functions in the organization have sufficient authority and organizational freedom to perform their critical functions effectively and without reservation.

The quality assurance program applies to all safety related structures, systems, and components within the Stone and Webster scope of work. Stone and Webster has also committed to comply with the NRC quality assurance guidance provided in the Gray Book and the Green Book.

The program for engineering includes quality assurance review of applicable engineering instructions, procedures, specifications, and drawings to assure the quality requirements are clearly, accurately, and adequately stated. The program requires that design work be verified or reviewed by individuals within the engineering organization not responsible for originating the design and that a determination is made that the engineering specifications, procedures, instructions, and drawings comply with regulatory requirements and design bases.

For procurement control, quality assurance measures provide for the review of procurement documents to assure that the stated quality requirements are adequate, for supplier qualification, and for approval of the suppliers' quality assurance programs. The program provides for inspection, surveillance, and audit of the suppliers' safety related structures, systems, and components to assure compliance with procurement requirements.

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During construction, the program provides for onsite quality assurance involvement including inspection, nondestructive testing, retention of records, and processing of deficiencies, nonconformances, and design changes. The quality control engineers, inspectors, and nondestructive testing personnel are organizationally separate and independent from the construction organization.

The program provides for a comprehensive system of detailed audits to be performed by the Stone and Webster organization. The audits encompass the review and evaluation of all quality related activities associated with the quality assurance program and involve procedures, work areas, hardware, activities, and records. The program requires that the audits be conducted in accordance with established procedures by qualified personnel not having direct responsibilities in the area being audited. The results are documented and distributed to the appropriate levels of management.

During our review, we evaluated the Stone and Webster quality assurance program for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the program contains the necessary quality assurance provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and is acceptable for the design, procurement, and construction of the Montague 1 and 2.

17.5 Implementation

The NRC Office of Inspection and Enforcement has concluded that the implementation of the overall quality assurance program cannot be fully assessed at the present time due to the pending revision of the applicants' quality assurance manual and the lack of current activity. This is an open item which will require resolution prior to initiation of safety related activities either under a Limited Work Authoriziation or prior to a decision for issuance of construction permits for Montague 1 and 2. This matter is discussed in Section 1.8 regarding schedule for completion and resolution.

17.6 Conclusion

We have reviewed and evaluated the quality assurance program of Northeast Utilities Service Company, General Electric and Stone and Webster for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review, we conclude that the quality assurance program is acceptable for the design, procurement, and construction of Montague 1 and 2. Prior to initiation of safety related activities under a Limited Work Authorization or prior to a decision for issuance of construction permits for Montague 1 and 2, the open item noted in Section 17.5 above must be resolved.

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18.0 REVIEW BY ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The application for the Montague 1 and 2 will be reviewed by the Advisory Committee On Reactor Safeguards. We intend to issue a supplement to this Safety Evaluation Report after the receipt of the Committee's report to the Commission relative to its review of the Montague application. The supplement will append a copy of the Committee's report and will address each of the comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The applicants state that the activities to be conducted will be within the jurisdiction of the United States and that all the directors and principal officers of all the applicants are citizens of the United States.

None of the applicants are owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicants have agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicants will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. We are continuing our review of the financial qualifications of the applicants and will report the results of our evaluations in a supplement to this report. As indicated in Section 1.8, our review of this matter will be resumed approximately one year prior to the applicants' scheduled date for beginning construction activities at the proposed site.

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21.0 CONCLUSIONS

Based on our evaluation of the proposed design for the Montague 1 and 2, we find that upon resolution of the outstanding matters set forth in Section 1.8 of this report and discussed in appropriate sections of this report, we will be able to conclude in accordance with the provisions of Section 50.35(a) of 10 CFR Part 50, that:

(1) The applicants have described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and have identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the final safety analysis report;

(3) Safety features or components which require research and development have been described and identified by the applicants, and there will be conducted research and development programs reasonably designed to resolve safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that (a) such safety questions will be satisfactorily resolved at or before that latest date stated in the application for completion of construction of the proposed facility, and (b) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The Northeast Nuclear Energy Company is technically qualified to design and construct the proposed facility;

(6) The applicants have reasonably estimated the costs and are financially qualified to design and construct the proposed facility; and

(7) The issuance of permits for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

CHRONOLOGY OF REVIEW

MONTAGUE 1 & 2

July 12, 1974	Application, Environmental Report, Preliminary Safety Analysis Report (Volumes 1-10) and Antitrust volume docketed. Docket Nos. 50-496/497 assigned to Montague Nuclear Station.
July 18, 1974	Applicants submitted supplemental information for Section 6.2 of the Preliminary Safety Analysis Report. This information consists of figures to be inserted into Volume 6 of the application.
July 25 & 26, 1974	AEC representatives meet with Stone & Webster and local geologists in Greenfield, Massachusetts to discuss and observe geological features around the Montague site.
July, 31, 1974	AEC/DL letter advising that application was docketed and stating which newspapers, trade journals and publications would publish notices.
August 2, 1974	Applicants submitted proprietary data and cites the GESSAR docket as a reference and requests that these tables and figures be withheld from public disclosure.
August 20, 1974	Applicants letter advising that it is planned to incorporate into the Montague Preliminary Safety Analysis Report a substantial amount of information directly from the GESSAR-238 Nuclear Island Standard Design application.
August 21, 1974	Applicants' letter requesting a Staff Management review of the technical and cost/benefit reasons cited in the General Electric letter of July 29, 1974.
August 29, 1974	AEC/DL letter establishing Montague review schedule.
August 30, 1974	Applicants submitted Amendment 2 to License Application and Supplement 1 to Preliminary Safety Analysis Report.
September 4, 1974	Meeting with officials of Town of Montague.

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September 5, 1974	Applicants' letter committing to the incorporation of all amendments to the GESSAR-238 Nuclear Island Design application for nuclear steam supply system.
September 11, 1974	Applicant's letter advising that an additional 6 months will be required to evaluate the General Electric topical report to be submitted to AEC on October 1, 1974, regarding anticipated transients without scram.
September 13, 1974	AEC/DL letter transmitting Round 1 Requests for Additional Information.
September 17, 1974	AEC and applicants' representatives meet to discuss the results of the geological investigation of the Montague Site.
September 30, 1974	Applicants' letter advising a one year delay for the Montague Nuclear Power Plant.
October 1, 1974	Representatives from AEC and applicants meet in Bethesda, Md. to discuss clarification of the load combinations set forth in Section 3.9.2.1A of the Montague application.
October 7, 1974	Notice and Order for Special Prehearing Conference issued by Atomic Safety and Licensing Board.
October 15, 1974	AEC/DL letter advising that General Electric documents submitted with the application for docketing have been withheld from public disclosure as proprietary information. These documents were previously submitted on the GESSAR-238 docket.
October 18, 1974	NNECo letter advising that A. Roisman, Esq., Counsel for the New England Coalition on Nuclear Pollution seeks to have the Montague Notice of Hearing withdrawn.
October 23, 1974	Applicants advise that Supplement No. 2 to the PSAR, to be filed as amendment, will be submitted by November 8, 1974.
October 24, 1974	AEC/DL letter transmitting additional information required regarding hydrology, geology and seismology matters.
October 30, 1974	Applicants advise that Supplement 2 to PSAR will be submitted by November 8, 1974 and Supplement 3 will be submitted by December 3, 1974.
November 8, 1974	Applicants submitted Amendment No. 5 to the Montague license application. This amendment consists of a volume entitled

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	Supplement No. 2 to the application. Included in this supplement are replacement pages which incorporate certain portions of Amendments 13 through 20 to the GESSAR-238 docket.
November 11, 1974	AEC/DL letter transmitting a change in the schedule based on the one year delay in start of construction.
November 25, 1974	Applicants' letter concurring with the schedule transmitted on November 11, 1974 by AEC/DL.
December 4, 1974	Order issued by the Atomic Safety and Licensing Board admitting the Commonwealth of Massachusetts as a party, and the New England Coalition on Nuclear Pollution as a party; the State of New Hampshire will participate as a state.
December 9, 1974	Applicants submitted site Meteorological Data and Aerial Photographs.
December 12, 1974	Applicants submitted Amendment No. 6 to Montague application. This amendment consists of Supplement 3 to Preliminary Safety Analysis Report and Supplement 4 to the Environment Report.
December 13, 1974	Applicants' letter transmitting certain figures contained in Section 7.6 which were omitted from Supplement 3 of Amendment 6 transmitted on December 12, 1974.
January 30, 1975	Order issued by Atomic Safety and Licensing Board. The Board requests memorandums of law by February 10, 1975 with regard to the letters from Commonwealth of Massachusetts, the county of Franklin and the State of New Hampshire requesting relief with regard to meetings which may be held between the applicants and the staff.
February 6, 1975	Applicants and NRC meet to discuss outstanding information regarding site geology.
February 19, 1975	NRC letter transmitting Round 2 Requests for Additional Information.
February 21, 1975	Applicants' letter transmitting Amendment No. 8 which consists of Supplement No. 4 to the Preliminary Safety Analysis Report.
February 19, 1975	Applicants' letter regarding anticipated transients without scram schedule and positions.

March 18, 1975	Applicants' letter containing responses to DRL questions concerning geology/seismology.
March 21, 1975	Applicants' letter concerning Outstanding Geology Information.
April 1, 1975	Order granting the petitions to intervene filed on behalf of County of Montague, Turners Falls Airport, Inc., and Pioneer Aviation Corporation.
April 1, 1975	Applicants' letter advising that the Board of Trustees voted to defer commercial operation of the Montague Station for a period not to exceed four years.
April 22, 1975	DRL letter concerning Mark III containment system.
April 23, 1975	Order issued by the Atomic Safety and Licensing Board. Order denies motion made by the Commonwealth of Massachusetts to hold all hearings in New England, pay the transportation of intervenor's counsel or provide transcripts.
May 2, 1975	Applicants submitted Amendment 10 consisting of Supplement No. 5. This supplement contains responses to DRL letters regarding geology.
May 2, 1975	Applicants letter advising that a supplement to the application will be submitted by July 3, 1975 addressing all Q-2 requests.
May 8, 1975	DRL transmits a revised review schedule for Montague.
May 8, 1975	Applicants' letter advising that Supplement No. 7 to the Environmental Report will be submitted by June 27, 1975.
May 23, 1975	Applicants' letter advising that they concur with the review schedule transmitted on May 8, 1975.
June 10, 1975	NRC and applicants' representatives meet to discuss responses to Round 2 Requests for Additional Information.
July 3, 1975	Applicants transmitted Amendment No. 12 - Supplement No. 6 to the application consisting of responses to outstanding requests for additional information.
July 8, 1975	DRL letter concerning emergency core cooling system - final acceptance criteria.

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July 17, 1975	Applicants' letter regarding emergency core cooling system - final acceptance criteria to be submitted by General Electric by the end of August 1975.
July 18, 1975	Applicants' letter transmitting monthly joint frequency distribution meteorology data.
July 30, 1975	Applicants' letter concerning meteorology calculations and X/Q factors to be used in accident analyses and plant design.
August 1, 1975	Applicants' letter containing responses to outstanding requests.
August 12, 1975	Representatives of NRC and applicants meet to discuss meteorology analysis performed for the accident dose calculations.
September 4, 1975	DRL letter concerning resolution of outstanding items before issuance of the Safety Evaluation Report. Radiological Review Schedule enclosed.
September 9, 1975	DRL letter requesting additional information in regard to Section 17.0 of the Preliminary Safety Analysis Report.
October 14, 1975	Applicants' letter transmitting commitments proposed to resolve outstanding items in staff's safety review.
November 12, 1975	Meeting held in Bethesda with applicants to discuss schedule for issuance of the staff's safety evaluation report.
November 25, 1975	Meeting with applicants held in Bethesda to discuss site meteorological data.
December 12, 1975	Amendment 14 Supplement 7 submitted addressing items contained in August 1, 1975 and October 12, 1975 letters.
January 22, 1976	Applicants' letter describing appropriate building design leak rates and pressure response following an accident.
April 1, 1976	NRC meeting with applicants to discuss resolution of outstanding items.
April 6, 1976	Letter sent to applicants concerning design of structures within and above the suppression pool.
April 15, 1976	NRC meeting with applicants to discuss the New England seismicity and geology as it affects the Montague site.

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April 20, 1976	Notice and Order for special prehearing conference issued.
April 21, 1976	Applicants' letter providing additional clarification of outstanding matters.
April 28, 1976	Applicants' letter providing clarification of quality assurance annual review to assess program effectiveness.
May 5, 1976	Applicants' letter transmitting New England seismic strain relief map discussed at April 15, 1976 meeting.
May 10, 1976	NRC letter to applicants concerning changes to 10 CFR Parts 2, 50 and 51.

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APPENDIX B

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Note: Documents referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the PSAR, may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicants and Commission's Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. Correspondence between the applicants and the Commission may also be inspected at the Carnegie Library, Avenue A, Turners Falls, Massachusetts. Specific documents relied upon by the NRC staff and referenced in this Safety Evaluation Report are as follows:

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