

SECTION 1 INTRODUCTION AND SUMMARY

TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
1.1	Purpose, Scope and Organization of Report	3
1.1.1	Introduction.....	3
1.1.2	Methods of Technical Presentation	4
1.1.2.1	Purpose	4
1.1.2.2	Radioactive Material Barrier Concept.....	4
1.1.2.3	Organization of Contents.....	5
1.1.2.4	Format Organization of Subsections	6
1.2	Principal Design Criteria.....	7
1.2.1	General Criteria	7
1.2.2	Reactor Core	7
1.2.3	Reactor Core Cooling	8
1.2.4	Plant Containment.....	8
1.2.5	Plant Instrumentation and Control	9
1.2.6	Plant Electrical Power.....	9
1.2.7	Plant Radioactive Waste Disposal.....	9
1.2.8	Plant Shielding and Access Control.....	9
1.2.9	Plant Fuel Handling and Storage.....	9
1.2.10	Separation of Safety Systems	10
1.2.11	Class I Equipment and Structures	10
1.2.12	Class II Equipment and Structures	10
1.3	Summary Design Description and Safety Analysis.....	10
1.3.1	Plant Site and Environs	10
1.3.1.1	Description of the Site	10
1.3.1.2	Description of the Environs.....	11
1.3.1.3	Geology	11
1.3.1.4	Hydrology	11
1.3.1.5	Regional and Site Meteorology	12
1.3.1.6	Seismology and Design Response Spectra	12
1.3.1.7	Environmental Monitoring Program	12
1.3.2	Reactor System	13

SECTION 1 INTRODUCTION AND SUMMARY

TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
1.3.3	Plant Containment System	14
1.3.3.1	Primary Containment System	14
1.3.3.2	Secondary Containment System	16
1.3.4	Plant Auxiliary and Standby Cooling Systems	16
1.3.5	Plant Instrumentation Control Systems	17
1.3.5.1	Reactor Control	17
1.3.5.2	Reactor Protection System	18
1.3.5.3	Plant Radiation Monitoring System	18
1.3.6	Plant Fuel Storage and Handling Systems	18
1.3.7	Plant Main Turbine System	19
1.3.8	Plant Electrical Power Systems	19
1.3.9	Plant Shielding, Access Control, and Radiation Protection Procedures	19
1.3.10	Plant Radioactive Waste Control Systems	20
1.3.11	Summary Evaluation of Plant Safety	21
1.3.11.1	General	21
1.3.11.2	Summary of Off-site Doses	21
1.4	Identification and Qualification of Contractors	21
1.4.1	Licensee	21
1.4.2	Contractors	21
1.5	References	22
FIGURES	24
Figure 1.3-1a	Normal Reactor Heat Balance 2004 MWt	25
Figure 1.3-1b	Increased Feedwater Temperature Reactor Heat Balance 2004 MWt	26
Figure 1.3-1c	Increased Feedwater Temperature Reactor Heat Balance 2004 MWt for MELLLA+/EFW	27
Figure 1.3-2	Turbine Generator System – Heat Balance	28
Figure 1.3-3	Plant Process Flow Diagram	29

SECTION 1 INTRODUCTION AND SUMMARY**1.1 Purpose, Scope and Organization of Report****1.1.1 Introduction**

This Updated Safety Analysis Report (USAR) is submitted by Northern States Power Company, a Minnesota corporation herein designated as NSPM, for the Monticello Nuclear Generating Plant, herein designated as the plant, in accordance with the requirements of 10CFR50 Section 50.71(e) as published in the Federal Register on May 9, 1980.

The Monticello Nuclear Generating Plant (MNGP) is owned by Northern States Power Company (NSP). NSP is a wholly owned utility operating subsidiary of Xcel Energy Corporation (Xcel Energy) (Reference 3). Transfer of operating authority for the plant from NSP to NMC was approved by the Nuclear Regulatory Commission (NRC) in License Amendment 110 (Reference 4). The operating authority was transferred back to NSPM as approved by the NRC in License Amendment 156 (Reference 5).

This USAR is the updated version of the Final Safety Analysis Report (FSAR). The FSAR was originally submitted on November 8, 1968. That document will be referred to as the "original FSAR". The last amendment to the FSAR (number 28) was submitted on July 23, 1970. Following July 23, 1970 the FSAR was not amended and became a historical document. This document will be referred to as the "FSAR". The USAR contains a current description of the Monticello Nuclear Generating Plant as of the latest revision date (see "Document Control" Section). This document will be revised per 10CFR50 Section 50.71(e).

The Monticello Nuclear Generating Plant, Unit 1, uses a single cycle, forced circulation, low power density boiling water reactor. General Electric Company designed the plant and supplied the nuclear steam supply system, the initial reactor fuel, and turbine-generator unit and its related systems. This design is identified as "BWR-3" by General Electric. Bechtel Corporation constructed the plant.

The plant was constructed, pursuant to Construction Permit CPPR-31, at the Monticello site in Wright County, Minnesota.

Construction started on June 19, 1967, and initial fuel loading was completed during the fall of 1970. Following a period of testing, full commercial operation began on June 30, 1971 under Provisional Operating License Number DPR-22. The Full Term Operating License was issued on January 9, 1981. The Monticello renewed license expires midnight September 8, 2030.

This USAR contains an analysis and evaluation of the plant, including the core, based on operation at 1670 MWt, which is equivalent to a nominal gross electrical output of 575.4 MWe. The Facility Description and Safety Analysis Report submitted in support of the application for Construction Permit CPPR-31 evaluated the design of the major systems and components of the plant, including the containment and engineered safeguards, at a power level of 1674 MWt. The thermal and hydraulic characteristics of the core, however, were evaluated at a power level of 1469 MWt, the then contemplated initial power level for which an operating license would be sought. Based on more recently developed critical heat flux correlations, the final fuel design, and the radioactivity release rates related to operation at 1670 MWt, authorization was

SECTION 1 INTRODUCTION AND SUMMARY

requested for operation of power levels up to 1670 MWt. The startup and test program set forth in Appendix D provided a stepwise increase in power levels up to 1670 MWt and the criteria which were met prior to proceeding to operation at each of the succeeding step increases in power level.

Approval to increase thermal power levels to 1775 MWt and subsequently to 2004 MWt were granted by the NRC in September 1998 (Reference 6) and December 2013 (Reference 7), respectively. Implementation of each of these power uprates involved a power ascension test program which took into account applicable elements of the original startup test program set forth in Appendix D. The power ascension program for the 2004 MWt uprate also included testing of the replacement steam dryer to verify that acoustic loads were within predicted limits. Summary report of the completed 1775 MWt test program was submitted in February 1999 (Reference 2). The 2004 MWt test results for the steam dryer were submitted to the NRC as required (Reference 14).

On March 28, 2014, the reactor operating domain was expanded to include the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region (Reference 13). When the MELLLA+ operating domain expansion was implemented, testing was performed to confirm operational performance and control aspects of the associated changes. This testing is described in Section 10.4 of the Safety Analysis Report for MELLLA+ (Reference 12).

In 2014, 2015, and 2017, NRC approval was given for use of a Spent Fuel Pool criticality analysis performed by AREVA, for use of AREVA fuel licensing methods and use of ATRIUM-10XM fuel in the core, and for core operation in a power-flow range called the Extended Flow Window (EFW), which is identical to the MELLLA+ operating domain. (References 15, 16, 17).

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1.1.2 Methods of Technical Presentation**1.1.2.1 Purpose**

This USAR contains the changes necessary to reflect significant information and analyses submitted to the Commission or prepared by NSPM or its predecessor, NMC, pursuant to Commission requirements since the submission of the original FSAR.

1.1.2.2 Radioactive Material Barrier Concept

Because the safety aspects of this report pertain to the relationship between plant behavior under a variety of circumstances and the radiological effects on persons off-site, the report is oriented to the radioactive material barrier approach. This orientation facilitates evaluation of the radiological effects of the plant on the environment and to the health and safety of the general public.

The overriding consideration that determines the depth of detailed technical information presented about a particular system or component is the relationship of the system or component to the radioactive material barriers. Systems that must operate to preserve the radioactive material barriers are described in the greatest detail. Systems that have little relationship to the radioactive material barriers are described only with as much detail as is necessary to establish their functional role in the plant.

SECTION 1 INTRODUCTION AND SUMMARY**1.1.2.3 Organization of Contents**

The USAR Master Table of Contents is provided with Volume One and shows 2-digit subsection detail only. Tables of Contents for each section are provided at the beginning of each section.

The USAR is organized into 15 major sections each of which consists of a number of subsections. Tables are included within and at the end of the subsection in which they are referred to. References are listed in a separate subsection, just ahead of the figures at the end of each major section. Some of the references used in the USAR may not have been previously submitted to the NRC, but most are available from NSPM. There are some instances where references were included in the USAR because they were listed in the FSAR, but a copy of the reference was not originally obtained from the vendor or are no longer available. These references are generally related to historical information and are not relied on as a basis for demonstrating the design adequacy of the MNGP.

The listing of effective pages showing the current revision of each subsection is provided with Volume One. To the maximum extent practicable, the current revision of each subsection of the appendices is also shown in Volume One; however, in some cases an appendix may be revised in its entirety at each revision and subsection revision control is not applicable.

The principal architectural and engineering criteria which define the broad frame of reference within which the plant is designed are set forth in subsection 1.2. Subsection 1.3 presents a brief description of the site environs and key plant systems.

Sections 2 through 13 present detailed information about the design and operation of the plant. In these sections nuclear safety systems and engineered safeguards are integrated into sections according to system function (emergency core cooling, control, etc.), system type (electrical, mechanical, etc.), or according to their relationship to a particular radioactive material barrier (primary containment, secondary containment, etc.). Section 3, "Reactor", describes plant components and presents design details that are most pertinent to the fuel barrier. Section 4, "Reactor Coolant System," describes plant components and systems that are most pertinent to the reactor system process barrier. Section 5, "Containment Systems", describes the primary and secondary containment systems. Thus, Sections 3, 4, and 5 represent the first four of seven plant radioactive material barriers.

The remainder of the sections group "system" information according to plant function (radioactive waste control, emergency core cooling, power conversion, control, etc.) or system type (electrical, structures, etc.). Subsections presenting information on topics other than plant systems or components are arranged individually according to the subject matter so that the relationship between the subject and public safety is emphasized.

Section 14, "Plant Safety Analysis," provides an overall safety evaluation of the plant which demonstrates both the adequacy of equipment designed to protect the radioactive material barriers and the ability of the safeguard features to minimize or mitigate the consequences of situations in which one or more radioactive material barriers are assumed damaged for analytical purposes.

SECTION 1 INTRODUCTION AND SUMMARY

Section 15, "USAR Drawings" provides a consolidation of drawings which are referred to in various subsections of the USAR.

The appendices to the FSAR describe and evaluate (a) the site and structures' seismic design criteria, (b) the site meteorology and limits, (c) quality assurance programs, (d) plant start-up program, (e) conformance to the AEC's proposed 70 design criteria, (f) the containment vessel design report, and (g) AEC questions and responses. A separate report describes fabrication, erection and testing of the reactor vessel. Each of these documents has been included in the USAR either by reference or actual text incorporation, as appropriate.

Incorporated into the design of this plant are features to improve both operational performance and overall safety which have been presented in special topical reports.

1.1.2.4 Format Organization of Subsections

Subsections are numerically identified by representing their order of appearance in a section by two numbers separated by a decimal point, e.g., 3.4, the fourth subsection in Section 3. Subsections are further subdivided by numbers separated by decimal points (3.4.1.1, etc.). Pages within each subsection are consecutively numbered.

Tabulations of data are designated "Tables" and are identified by the subsection number followed by a hyphen and the number of the table, e.g., Table 3.4-5. Pictures, sketches, curves, graphs, and engineering diagrams are identified as "Figures" and are numbered in the same manner as tables. Drawings are referred to by NSP drawing number and are contained in Section 15.

The general organization of a subsection describing a system or component usually follows:

- a) Design Basis
- b) Description
- c) Performance Analysis
- d) Inspection and Testing

SECTION 1 INTRODUCTION AND SUMMARY**1.2 Principal Design Criteria**

The principal criteria for design, construction, and operation of this plant are summarized below.²

1.2.1 General Criteria

- a. The plant is designed, fabricated, erected, and operated to produce electrical power in a safe, reliable and efficient manner and in accordance with applicable codes and regulations.
- b. The plant is designed in such a way that the release of radioactive materials to the environment is limited, so that the limits and guideline values of published regulations pertaining to the release of radioactive materials are not exceeded.
- c. The design of those components which are important to the safety of the plant includes allowances for the appropriate environmental phenomena at the site. Those components important to safety and required to operate during accident conditions are designed to operate in the post accident environment.

1.2.2 Reactor Core

- a. The reactor core is designed as a boiling water reactor to produce steam for direct use in a turbine-generator.
- b. The reactor core, in conjunction with other design parameters, is designed so there is no inherent tendency for sudden divergent oscillation of operating characteristics in any mode of operation.
- c. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.
- d. Power excursions which could result from any credible reactivity addition accident do not cause damage, either by motion or rupture, to the reactor vessel or impair operation of required safeguards.
- e. The reactor core is designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the reactor core subcritical and maintaining it so from any power level in the operating cycle.
- f. Redundant backup reactivity shutdown capability is provided independent of normal reactivity provisions. This system has the capability, with adequate margin, to shut down the reactor from any power level in the operating cycle.
- g. The fuel rod cladding is designed to contain the fission gas released from the fuel material throughout the design life of the fuel rod.

2. For a comparative evaluation of the Monticello plant with the AEC's proposed 70 General Design Criteria, refer to Appendix E of the USAR.

SECTION 1 INTRODUCTION AND SUMMARY

- h. Thermal characteristics of the reactor core are adequate to prevent fuel clad surface heat flux or fuel material center temperatures which could cause sudden fuel cladding ruptures.
- i. The reactor core and associated systems are designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

1.2.3 Reactor Core Cooling

- a. Heat removal systems are provided to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to maximum thermal output. The capacity of such systems is adequate to prevent fuel clad damage.
- b. Heat removal systems are provided to remove decay heat generated in the reactor core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel clad damage.
- c. Redundant heat removal systems are provided to preserve reactor core heat transfer geometry following various postulated design basis loss-of-coolant accidents.
- d. Independent means are provided to prevent overpressure conditions which could jeopardize the integrity of the reactor primary system or reactor core cooling systems.

1.2.4 Plant Containment

- a. The primary containment system is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture, or equivalent failure of any coolant pipe within the primary containment.
- b. Provision is made both for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain the integrity of the primary containment system as long as necessary following the various postulated design basis loss-of-coolant accidents.
- c. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides for primary containment when the primary containment system is open and the containment function is required.
- d. Provisions are made for preoperational pressure and leak rate testing of the primary containment system and for leak testing at periodic intervals. Provision is also made for leak testing selected penetrations. Provision is also made for demonstrating the functional integrity of the secondary containment system.

SECTION 1 INTRODUCTION AND SUMMARY

- e. The integrity of the complete plant containment system and such other associated engineered safeguards as may be necessary are designed and maintained so that offsite and Control Room operator doses resulting from postulated design basis accidents are below the values stated in 10CFR50.67.

1.2.5 Plant Instrumentation and Control

- a. The plant is provided with a main control room having adequate shielding and air conditioning facilities to permit occupancy for normal plant operation as well as during all postulated design basis accident situations.
- b. Interlocks or other protective devices are provided so that procedural controls are not the only means of preventing serious accidents.
- c. A reliable reactor protection system, independent from the reactor process control system, is provided to automatically initiate appropriate action whenever plant conditions approach pre-established limits. Periodic testing capability is provided. Sufficient redundancy is provided so that failure or removal from service of any one component or portion of the system will not preclude appropriate actuation of the reactor protection system when required.

1.2.6 Plant Electrical Power

Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safeguards functions under all postulated design basis accident conditions.

1.2.7 Plant Radioactive Waste Disposal

- a. Gaseous, liquid and solid waste disposal systems are designed so that discharge of effluents and off-site shipments will be in accordance with 10CFR20 and other applicable regulations.
- b. Process and discharge streams are appropriately monitored and such features incorporated as may be necessary to maintain releases below the permissible limits of 10CFR20.

1.2.8 Plant Shielding and Access Control

The radiation shielding in the plant and the plant access control patterns are such that the personnel doses are as low as reasonably achievable and well below the limits of 10CFR20.

1.2.9 Plant Fuel Handling and Storage

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide cooling for the spent fuel.

SECTION 1 INTRODUCTION AND SUMMARY**1.2.10 Separation of Safety Systems**

Systems and equipment provided for the prevention of and the mitigation of the consequences of accidents are provided in such redundancy and physical separation that the accident will not preclude operation of sufficient equipment to effectively control the effects of the accident.

1.2.11 Class I Equipment and Structures

Class I structures, systems and components are those whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant under normal or accident conditions and to the removal of decay and sensible heat from the reactor.

1.2.12 Class II Equipment and Structures

Class II structures, systems and components are those whose function is not vital or essential to safe shutdown.

1.3 Summary Design Description and Safety Analysis**1.3.1 Plant Site and Environs**

Section 2 provides detailed information on the site and environs of the Monticello Nuclear Generating Plant which confirms the suitability of the site. This section summarizes the principal design characteristics of the site and environs.

1.3.1.1 Description of the Site

The plant is located within the city limits of Monticello, Minnesota (1990 population 4,941), on the south bank of the Mississippi River in Section 33, T-122N, R-25W, in Wright County, Minnesota, at 45° 20' N latitude and 93° 50' W longitude. Approximately 2150 acres of land are owned in fee by Northern States Power Company (NSP) at the plant location. NSP is a wholly owned utility operating subsidiary of Xcel Energy Corporation (Xcel Energy) (Reference 3). The property is divided by the river with part being in Sherburne County and part in Wright County. Drawing ND-95208, Section 15, shows the Monticello property map.

The immediate plant area, including major portions of the intake, is completely enclosed by a double security fence. Access to the plant is through the Security Access Facility or Security Gate. The access road extends from the security gate to County Road 75, 3000 feet southeast of the reactor building. Interstate Highway #94 is located 3700 feet southwest. Railroad access is provided by the Burlington Northern Railroad. Air access is provided by the Twin Cities Airport of Minneapolis - St. Paul, located approximately 45 miles southeast of the site.

SECTION 1 INTRODUCTION AND SUMMARY**1.3.1.2 Description of the Environs**

The area around the site is used for agriculture. The nearest house to the reactor building is about 0.6 miles southwest. The nearest well serving more than one home is located in the city of Monticello. The city, which consists of a small commercial complex and attendant residential development, includes the Wright County portion of the plant site within its boundary. The resident population within the 10 mile Emergency Planning Zone (EPZ) (2010 estimate) is approximately 68,635. The resident population within the 5-mile EPZ (2010 estimate) is 45,447. The northwestern suburbs of Minneapolis are about 30 miles from the site.

From the population and land usage viewpoint, it is concluded that the site is suitable for the plant, considering the containment and additional engineered safeguards provided as an integral part thereof.

1.3.1.3 Geology

The site area is covered by unconsolidated deposits of dense soils underlain by Paleozoic sedimentary rock at about 75 to 122 feet. The reactor building foundation is founded on a layer of compacted granular backfill overlying a dense sand and gravel layer which covers a fine to medium grain sandstone. The turbine building foundation rests on a dense layer of sand and gravel on which the reactor backfill is founded. Seismic surveys disclosed no unusual or extreme subsurface conditions.

The geology of the area and soil tests indicate that the rock and soil loading capacity is adequate to support the reactor building and related structures.

1.3.1.4 Hydrology

The finished plant grade is about 25 feet above mean river level, (905 MSL), 14 feet above the record flood (916 feet MSL-1965), and 10 feet above the predicted 1,000 year flood.

The "probable maximum flood" criterion as defined by the U.S. Corps of Engineers was used to establish the maximum flood level. Using this criterion, the flood analyses predicted a probable maximum flood peak stage at the site of approximately nine feet above plant grade. The peak level at the site would be reached in about 12 days from the onset of the worst combination of hydrometeorological, hydrological and climatic conditions resulting in the probable maximum flood.

River flows vary widely throughout the year. Generally maximum flows occur in the spring, and minimum flows occur in late summer (July, August, September) or mid-winter (January, February). The low flow of record is 220 cfs. The mean flow is 3400 cfs and the average flow is 4600 cfs. The plant design and construction (including radioactive waste control systems) and contingency procedures take into consideration the extremes of river flow and stage (i.e., the probable maximum flood).

SECTION 1 INTRODUCTION AND SUMMARY**1.3.1.5 Regional and Site Meteorology**

The meteorology of the site area is basically that of a continental location with its associated favorable atmospheric dilution conditions prevailing. Diffusion climatology comparisons with other locations indicate that the site is typical of the midwestern United States. Inversion conditions exist at the site approximately 30 to 40% of the time.

All Class I and II structures are designed to withstand the maximum potential loadings resulting from a wind speed of 100 mph at 30 feet above ground with a gust factor of 1.1. The design is in accordance with standard codes and normal engineering practices.

It is estimated that the probability of experiencing tornadic forces at the site is of the order of one chance per 2000 years. In spite of this low probability, the plant is designed so that features of the plant important to continuity of reactor core cooling are designed to withstand the forces of short term tornado loadings of 300 mph.

1.3.1.6 Seismology and Design Response Spectra

The seismic design for critical structures and equipment for this plant is based on dynamic analysis of acceleration or velocity response spectrum curves which are based on a horizontal ground acceleration of 0.06 g.

The natural periods of vibration are calculated for buildings and equipment which are vital to the safety of the plant. Damping factors are based upon the materials and methods of construction used.

Earthquake design is based on ordinary allowable stress as set forth in the applicable codes and is very conservative because the usual one-third increase in allowable working stresses due to loadings from the operating basis earthquake is not used. As an additional requirement, the design is such that a safe shutdown can be made following a safe shutdown earthquake assuming a horizontal ground acceleration of 0.12 g.

The 0.12 g design criteria are for critical items only; that is, for Class I items.

1.3.1.7 Environmental Monitoring Program

An environmental radiation monitoring program was initiated in 1968 prior to the start of plant operation and continued after plant operation began.

The current Radiological Environmental Monitoring Program (REMP) is a comprehensive program of sampling and analysis of the air, terrestrial, and aquatic environments for radioactivity. The types of samples and sample locations included in the current REMF at Monticello are specified in the Off-site Dose Calculation Manual (ODCM) (Reference 1).

SECTION 1 INTRODUCTION AND SUMMARY**1.3.2 Reactor System**

The reactor is a single-cycle, forced circulation, low power density boiling water reactor producing steam for direct use in a steam-turbine. The reactor core includes the fuel assemblies and control rods.

The reactor core is assembled in modules of four fuel assemblies set in the interstices of a cruciform control rod. This modular core form, common to all General Electric boiling water reactors, permits substantial increase in thermal power with a small increase in core diameter and at the same time preserves the reactivity control characteristics.

The reactor vessel contains the reactor core and supporting structure, steam separator and dryer assemblies, jet pumps, control rod guide tubes, and the Reactor Feedwater, Emergency Core Cooling System (ECCS), and Standby Liquid Control System spargers and other components as shown in Figure 3-61. The inside diameter of the reactor vessel is approximately 17-feet 1-inch and the inside height between top head and bottom cap is approximately 63-feet 2-inches. Some of the main connections to the reactor vessel include the four main steam lines, two jet pump motive flow recirculation loop lines, four reactor feedwater lines, ten jet pump inlet lines, and one hundred and twenty-one control rod drive thimbles. Other connections are provided for the reactor Standby Liquid Control System, ECCS, and the various instrumentation and control systems.

The fuel for the reactor core consists of slightly enriched uranium dioxide pellets contained in sealed Zircaloy tubes. These fuel rods are assembled into individual fuel assemblies with a 100 rod array. Not all positions in this array are occupied by fuel rods. Each fuel assembly is fitted with a Zircaloy flow channel. Water serves as both the moderator and coolant for the core. The complete core loading consists of 484 fuel assemblies.

Control of reactivity is accomplished through control rod movements. The control rods are of cruciform shape and are dispersed throughout the lattice of fuel assemblies. The control rods are of the bottom-entry type and are moved vertically within the reactor core by individual, hydraulically operated, locking piston type control rod drives.

The Control Rod Drive System is designed to allow control rod withdrawal or insertion at a limited rate, one control rod at a time, for power level control and flux shaping during reactor operation. Stored energy available from gas-charged accumulators and/or from reactor pressure provides hydraulic power for rapid simultaneous insertion of all control rods for rapid (scram) reactor shutdown. Each control rod has its own separate drive mechanism, control, and scram devices.

The operational reactivity control system is of the same design as those used in other General Electric designed reactors. Temporary control curtains fabricated of boron stainless steel were installed between fuel channels during early life of the initial core to supplement the reactivity control of the control rods.

Reactor coolant enters the bottom of the reactor core and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separator and dryer assemblies located within the reactor vessel. The steam passes through main steam lines to the turbine. The separated water mixes with the incoming feedwater and is returned to the core bottom inlet through jet pumps located

SECTION 1 INTRODUCTION AND SUMMARY

within the reactor vessel. The motive force for the jet pumps is supplied by the water from the two Reactor Recirculation System loops. Each loop has a variable speed centrifugal pump with mechanical seals, motor operated gate valves for isolation of pumps for maintenance, and instrumentation for recirculation flow measurements and control.

Heat balances for the reactor system are shown on Figures 1.3-1a and 1.3-1b. The heat balance in Figure 1.3-1a is based on several parameters that utilize a nominal or representative value for input rather than specifically calculated values. The use of these nominal values was determined to be acceptable based on engineering judgment. The use of these nominal values has a negligible effect on the results of the reactor heat balance and the results provided were used as inputs for various EPU evaluations and Technical Specifications. The heat balance in Figure 1.3-1b is based on the same input values as Figure 1.3-1a except a higher feedwater heater performance and higher RWCU flow were assumed. The feedwater performance in the heat balance was adjusted to reflect actual higher than predicted feedwater temperatures at 1775 MWt resulting from the new feedwater heaters installed during the 2011 refuel outage. The higher feedwater temperatures delineated in this heat balance were evaluated for EPU (References 7, 8, 9, 10, and 11).

Figure 1.3-1c provides a reactor heat balance for higher feedwater heater performance under MELLLA+/EFW core flow conditions.

1.3.3 Plant Containment System

The Primary Containment System, consisting of a steel lightbulb-shaped drywell, a steel doughnut-shaped pressure suppression chamber, and interconnecting vent pipes, provides the first containment barrier surrounding the reactor vessel and reactor primary system. Any leakage from the Primary Containment System is to the Secondary Containment System which consists of the reactor building, the plant Standby Gas Treatment System, and the plant main stack. The integrated plant containment system and its associated engineered safety features are designed so that off-site and Control Room operator doses resulting from postulated design basis accidents are well below the reference values stated in 10CFR50.67.

1.3.3.1 Primary Containment System

The primary containment is designed to accommodate the pressures, temperatures, and hydrodynamic loads which would result from, or occur subsequent to a postulated loss-of-coolant accident (LOCA) within the primary containment and safety relief valve operations. The LOCA conditions evaluated in the primary containment design include the Design Basis Accident (DBA); that is, the pipe failure equivalent to a double-ended, circumferential rupture of a Reactor Recirculation System line resulting in the loss of reactor water at the maximum rate. The pressure suppression chamber is a steel, torus-shaped pressure vessel approximately half filled with water, and located below and encircling the drywell. The vent system from the drywell terminates below the water level of the pressure suppression chamber so that in the event of any pipe failure in the drywell, the released steam would pass directly to the water where it would be condensed. A bellows assembly connecting the suppression chamber to the vent line allows for differential movement between the drywell and the suppression chamber.

SECTION 1 INTRODUCTION AND SUMMARY

Isolation valves are provided on piping, penetrating the drywell and the suppression chamber, to provide integrity of the containment when required. These valves are actuated automatically by signals received from the containment isolation system. The valves on the auxiliary process systems are left open, or are closed, depending upon the functional requirements of that system, without reducing the integrity of the primary containment system.

Two features are included in the primary containment design to aid in maintaining the integrity of the Primary Containment System in the event of a postulated design basis loss-of-coolant accident. The containment spray cooling mode of the Residual Heat Removal System (RHR) provides redundant cooling capability for the removal of heat within the drywell and the pressure suppression chamber. Capability is provided in the containment structure design to withstand the forces exerted in the event that it is necessary to flood the primary containment vessel (drywell and suppression chamber) to a level which would flood the reactor core.

Each safety relief valve discharge line is routed to a standard Mark I T-quencher discharge device located in the suppression chamber below the water level. The T-quencher discharge device is used to ensure stable steam condensation at expected pool temperatures, and to mitigate pressure, thrust and hydrodynamic loads on the safety relief valve discharge line piping and the suppression chamber resulting from safety relief actuations.

After complete installation of all penetrations in the drywell and suppression chamber, these vessels were pressurized to the design pressure and measurements taken to verify that the integrated leakage rate from the integral vessels did not exceed the design leakage rate.

All containment closures which are fitted with resilient seals or gaskets are separately testable at pressures up to and including the containment design pressure to verify leak tightness. The covers on flanged closures, such as the equipment access hatch cover, the drywell head and access manholes are provided with double seals and with a test tap which allows pressurizing the space between the seals without pressurizing the entire primary containment system. Similarly, the space between the dual air lock doors can be pressurized to full design pressure.

Electrical penetrations have been provided with double seals and can be separately tested at pressures up to and including the containment design pressure. Electrical penetration assembly (EPA) X-101a is provided with a single aperture seal and double conductor seals. Double aperture seal is provided by a leak chase channel that monitors the EPA nozzle weld. The test taps and the seals are located so that the tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber.

Those pipe penetrations which must accommodate thermal movement are provided with expansion bellows-type seals. The bellows expansion joints are designed for the primary containment system design pressure and can be checked for leak tightness when the Primary Containment System is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized up to and including the containment design pressure to permit testing the individual penetrations for leakage.

SECTION 1 INTRODUCTION AND SUMMARY**1.3.3.2 Secondary Containment System**

Secondary containment is a controlled volume within the Reactor Building. The primary safeguards functions of the secondary containment are to minimize ground level release of airborne radioactive materials, and to provide for controlled, filtered, elevated release of the secondary containment atmosphere under postulated design basis accident conditions. Most of the Reactor Building is part of the secondary containment, and the Reactor Building provides the structural integrity of the secondary containment.

A plant Standby Gas Treatment System is provided to filter the secondary containment ventilation exhaust and discharge it to the off-gas vent stack during plant secondary containment system isolation conditions.

For the design basis fuel handling accident, analysis using Alternative Source Term methodology has demonstrated that secondary containment integrity and operation of the Standby Gas Treatment System are not required to maintain offsite and Control Room operator doses below 10CFR50.67 and GDC 19 limits.

1.3.4 Plant Auxiliary and Standby Cooling Systems

In addition to the turbine-generator and the main condenser systems, multiple, independent, auxiliary process systems are provided for the purpose of cooling the reactor and primary containment system under various normal and abnormal conditions.

- a. A Reactor Core Isolation Cooling System (RCIC) is provided for a continuous supply of makeup and cooling water to the reactor core when the reactor is isolated from the turbine and when the normal feedwater systems are not available.
- b. A two loop Core Spray System (CS) is provided. The system is designed to pump water from the suppression chamber pool directly to the reactor core through spargers mounted in the reactor vessel above the active core in a manner which will prevent fuel clad damage after depressurization following postulated design basis loss-of-coolant accident.
- c. A residual heat removal system (RHR) is provided which serves the following functions:
 - (1) To inject water into the reactor vessel after depressurization following a postulated design basis loss-of-coolant accident in order to rapidly reflood the core and prevent fuel clad melting. (This is the Low Pressure Coolant Injection System (LPCI) mode of RHR.)
 - (2) To remove heat from the water in the suppression chamber pool. (This is the containment cooling mode of RHR.)
 - (3) To spray water into the drywell and torus as an augmented means of removing energy from the drywell as required subsequent to a postulated design basis loss-of-coolant accident. (This is the containment spray mode of RHR.)

SECTION 1 INTRODUCTION AND SUMMARY

- (4) To remove decay heat and sensible heat from the reactor primary system so that the reactor can be shut down for a refueling and service operation. (This is the reactor shutdown cooling mode of RHR.)
- d. A High Pressure Coolant Injection System (HPCI) is provided for removal of decay heat and to provide coolant inventory control and heat dissipation from the core to the suppression chamber to prevent fuel clad damage following a postulated small break loss-of-coolant accident.
- e. An Automatic Depressurization System (ADS) is provided which, together with the core spray or the LPCI mode of the RHR will prevent fuel clad damage following a postulated small break loss-of-coolant accident.
- f. A standby coolant supply system is provided by a cross-tie between the Plant Service Water System and the Feedwater System which makes available an inexhaustible supply of cooling water from the river to the reactor core and containment independent of all other cooling water sources.
- g. An intertie is provided between the RHR Service Water System and the RHR discharge line. The plant Fire Protection System is also capable of discharging into this intertie line. This intertie provides an inexhaustible source of river water to the reactor core. A manual valve and blank flange is provided on the intertie to allow the connection to a source of water following a Beyond Design Basis Event.

The core cooling provisions itemized above (except f.) are designed to prevent fuel clad damage for the full range of primary system pipe size ruptures which may be postulated to occur without reliance upon off-site sources of power.

1.3.5 Plant Instrumentation Control Systems**1.3.5.1 Reactor Control**

Reactor power is controlled by movement of control rods and by regulation of the reactor coolant recirculation system flow rate. Control rods are also used to shape the reactor core power distribution. Procedural controls backed up by protective devices (reactor protection system, etc.) are used so that reactor core thermal performance does not exceed pre-established limits.

Reactor steam flow is automatically controlled by the Main Steam Pressure Control System which adjusts and controls steam flow to the main turbine in response to turbine inlet pressure. As a result, the plant turbine-generator power output follows the reactor power output.

A main turbine bypass system, having a capacity of approximately 11.5% of steam flow at rated load, is supplied with the turbine-generator system to restrict overpressure transients resulting from sudden complete or partial closure of the main turbine control valves or stop valves and provides a means of releasing steam to the main condenser during shutdown operations. The main turbine bypass system valves are operated on an overpressure signal from the Main Steam Pressure Control System. Rapid partial load rejection (up to 11.5% of rated flow) can be accommodated with the main turbine bypass system.

SECTION 1 INTRODUCTION AND SUMMARY**1.3.5.2 Reactor Protection System**

A Reactor Protection System is provided which automatically shuts down the reactor whenever the plant parameters monitored by the system approach pre-established limits.

The Reactor Protection System consists of two separately powered trip systems, Channel A and Channel B, each made up of two subchannels. The protection system receives inputs from sensors monitoring plant parameters. Each subchannel receives an input from at least one independent sensor monitoring each of the parameters. An unbypassed trip occurring in either subchannel (or both) of logic channel A, together with a unbypassed trip occurring in either sub-channel (or both) of logic channel B results in the opening of the scram valves in the Control Rod Drive System causing rapid insertion (scram) of the control rods. The Reactor Protection System is designed to cause a scram on loss of power to the Reactor Protection System. Components of the Reactor Protection System can be removed from service for testing and maintenance without interrupting plant operations and without negating the ability of the Reactor Protection System to perform its protective functions upon receipt of appropriate signals.

1.3.5.3 Plant Radiation Monitoring System

Instrumentation is provided for continuous monitoring of the radioactivity of specified process systems. Process systems where significant amounts of radioactivity may be present are monitored for any variation from normal. Certain nonradioactive processes are monitored to provide an alarm in the event they become contaminated due to the failure of a radiation barrier.

1.3.6 Plant Fuel Storage and Handling Systems

The refueling procedure is generally referred to as "wet" refueling since all irradiated fuel is always kept under water. The wet refueling procedure allows visual control of operations at all times. This feature is instrumental in producing a safe, efficient refueling sequence.

Spent fuel discharged from the reactor is transferred under water through the spent fuel storage pool canal into storage racks provided in the storage pool. The spent fuel storage pool is designed to accommodate the channel stripping operation and the many other fuel maintenance and inspection operations that are required. The spent fuel racks are designed and arranged so that the risk of criticality is eliminated. Storage space is also provided in the pool for irradiated fuel assembly channels and replaced control rods, the spent fuel cask, and the certain small internal components of the reactor core.

New fuel is brought through the equipment entrance of the Reactor Building and hoisted to the upper floor utilizing the reactor building crane.

Monticello has elected to comply with the criticality accident requirements of 10CFR50.68 in lieu of 10CFR70.24 for the handling and temporary storage of new fuel and non-fuel special nuclear material. Plant equipment and procedures have been evaluated and found to meet the requirements specified in 10CFR50.68 for precluding criticality events outside of the reactor.

SECTION 1 INTRODUCTION AND SUMMARY**1.3.7 Plant Main Turbine System**

The saturated steam leaving the reactor vessel flows through the four main steam lines to the main turbine located in the turbine building. After passing through the main turbine, the low pressure steam is condensed, the noncondensable gases are removed, and the condensate is demineralized before being returned to the reactor vessel through the Reactor Feedwater System heaters. Heat balances for the plant turbine system including the extraction steam subsystem are shown on Figure 1.3-2. A simplified process flow diagram for the entire plant is shown on Figure 1.3-3.

1.3.8 Plant Electrical Power Systems

The electrical output of the plant is fed into the plant site high voltage switchyard, and from the yard to Xcel Energy's network grid system via independent transmission lines (three 345 KV, two 230 KV and three 115 KV transmission lines). Plant Auxiliary electrical power is supplied from the 115 KV lines, and/or from the 345 KV switchyard. The plant Emergency Diesel Generator System (2 essential and 1 non-essential units) provides on-site standby emergency auxiliary electrical power.

The plant DC battery system consists of two 125 Vdc and three 250 Vdc batteries and systems which provide for controls and instrumentation which are vital to reactor and overall plant safety and to power certain functional requirements for reactor shutdown. Two separate 24 Vdc battery systems supply the Nuclear Instrumentation System and process radiation monitoring system.

1.3.9 Plant Shielding, Access Control, and Radiation Protection Procedures

Control of radiation exposure of plant personnel and people external to the plant is accomplished by a combination of radiation shielding, control of access into certain areas, and administrative procedures. The requirements of 10CFR20 were used as a basis for establishing the basic criteria and design bases.

Shielding is used to reduce radiation dose rates in various parts of the plant to acceptable limits consistent with operational and maintenance requirements. Access control and administrative procedures are used to limit the integrated dose received by plant personnel as low as reasonably achievable (ALARA). Access control and procedures are also used to limit the potential spread of contamination from various areas, particularly areas where maintenance occurs. The table below summarizes the design bases for shielding to assure that radiation levels in various areas of the plant are consistent with operational requirements.

SECTION 1 INTRODUCTION AND SUMMARYDesign Bases Plant Shielding Requirements

<u>Degree of Access Required</u>	<u>Design Dose Rate (DDE, mrem/hr)</u>
a) Continuous Occupancy	
Outside Controlled Access Area	0.5
Inside Controlled Access Area	1
b) Occupancy to 10 hr/week	6
c) Occupancy to 5 hr/week	12

The above design bases are at the shield walls. Generally, areas away from a shield wall receive lesser dose rates and this plus occupancy factors reduces the integrated dose received. Personnel involved in all phases of operation and maintenance normally receive far less than the permissible dose.

Both operating and shutdown conditions were considered in establishing the shielding design.

Shielding is also used as necessary to protect equipment from radiation damage. Of principal concern are organic materials such as insulation, linings, and gaskets.

1.3.10 Plant Radioactive Waste Control Systems

A Gaseous Radwaste System is provided to control, recombine, filter, store, monitor, and record the process off-gases as appropriate before release through the main plant stack during normal and abnormal plant operation.

A Liquid Radwaste System is provided for control, collection, treatment, storage, and disposal of liquid wastes. Liquid wastes are collected in sumps and drain tanks and transferred to the radwaste facility for further treatment, storage, or disposal.

In the Liquid Radwaste System, liquid wastes to be discharged from the system are handled on a batch basis with each batch being analyzed and disposed of as required. The system is designed to permit the waste to either be returned to the plant condensate system or be released to the Mississippi River after dilution in the discharge canal.

A Solid Radwaste System functions so that solid wastes are treated, sorted, packaged, solidified (if necessary), and shipped off-site.

SECTION 1 INTRODUCTION AND SUMMARY**1.3.11 Summary Evaluation of Plant Safety****1.3.11.1 General**

The general safeguards objectives of the design of this plant are to protect the equipment and to prevent radiation exposures in excess of a small fraction of pre-established limits to any persons on or off the plant premises, either during normal operation or during credible or postulated design basis accident conditions.

In order to meet these objectives, the design and operation include the following:

- a. Means for positive control of plant process parameters important to safety.
- b. Inherent safety features and automatic devices are included in the design to prevent a reactor operator error or equipment malfunction from causing an accident. Tests are conducted periodically to assure proper functioning of such devices.
- c. Multiple barriers are provided to contain the radioactive materials. The reactor core is conservatively designed to operate with thermal parameters significantly below those which could lead to fuel damage.
- d. The plant operating personnel are thoroughly knowledgeable in the operating characteristics of the plant, and are trained to follow written procedures to minimize the occurrence of operating errors.

1.3.11.2 Summary of Off-site Doses

The plant radioactive waste control systems for normal operation are designed to limit the radiation exposure of the off-site neighboring population to within the design objectives of Appendix I to 10CFR50.

1.4 Identification and Qualification of Contractors**1.4.1 Licensee**

By license amendment 156, dated September 22, 2008, the NRC made Northern States Power Company, a Minnesota corporation (NSPM) the licensee authorized to operate Monticello Nuclear Generating Plant (MNGP) (Reference 5)

1.4.2 Contractors

The Monticello Nuclear Generating Plant, Unit 1, was designed and built by General Electric Company as prime contractor for Northern States Power Company. General Electric engaged the services of the Bechtel Corporation as architect-engineer to provide the non-nuclear design and as engineer constructor for the plant.

SECTION 1 INTRODUCTION AND SUMMARY

Preoperational testing of equipment and systems and initial operation were performed by Northern States Power Company personnel with the technical assistance of General Electric. The initial staff for the Monticello Plant was drawn largely from the experienced staff of the Pathfinder Atomic Power Plant which Northern States Power Company operated. In addition, the initial staff had undergone extensive training during the construction phase of the plant. The plant was turned over to Northern States Power Company and responsibility for operation was assumed by NSP following demonstration of the operational capability at the contract specified output.

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy, entered into a Nuclear Power Plant Operating Services Agreement with NSPM. In accordance with that contract, NSPM has assumed exclusive responsibility for the operation and maintenance of the Monticello Nuclear Generating Plant (Reference 5).

1.5 References

1. Monticello Nuclear Generating Plant, Offsite Dose Calculation Manual (ODCM).
2. NSP (M F Hammer) letter to the NRC, "Power Rerate Test Program Startup Report," dated February 18, 1999.
3. NRC (C F Lyon) letter to NSP (W H Brunetti), "Issuance of Conforming Amendment re: Transfer of Facility Operating License from Northern States Power Company to a New Utility Operating Company Subsidiary (TAC No. MA7003)," dated August 18, 2000.
4. NRC (C F Lyon) letter to NMC (M B Sellman), "Issuance of Conforming Amendment re: Transfer of Operating Authority Under the Facility Operating License from Northern States Power Company to Nuclear Management Company, LLC (TAC No. MA7313)," dated August 7, 2000.
5. NRC (P S Tam) letter to NMC (T J O'Connor), "Issuance of Amendment to Conform the Renewed Facility Operating License to Reflect Transfer of Operating Authority (TAC No. MD9621)," dated September 22, 2008
6. NRC (T. Kim) letter to NSP (R O Anderson), "Issuance of Amendment RE: Power Uprate Program (TAC No. M96238)," dated September 16, 1998.
7. NRC (T A Beltz) letter to NSPM (K D Fili), "Monticello Nuclear Generating Plant - Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate (TAC No. MD9990)," dated December 9, 2013.
8. Monticello calculation 11-181, "Task Report T0100 - Reactor Heat Balance."
9. GE Hitachi Report NEDC-33322P, Revision 3, "Safety Analysis Report for Monticello Constant Pressure Uprate," October 2008.
10. NSPM letter L-MT-12-114 (M A Schimmel) to NRC, "Monticello Extended Power Uprate (EPU): Supplement for Gap Analysis Updates (TAC MD9990)," Item 10, dated January 21, 2013.

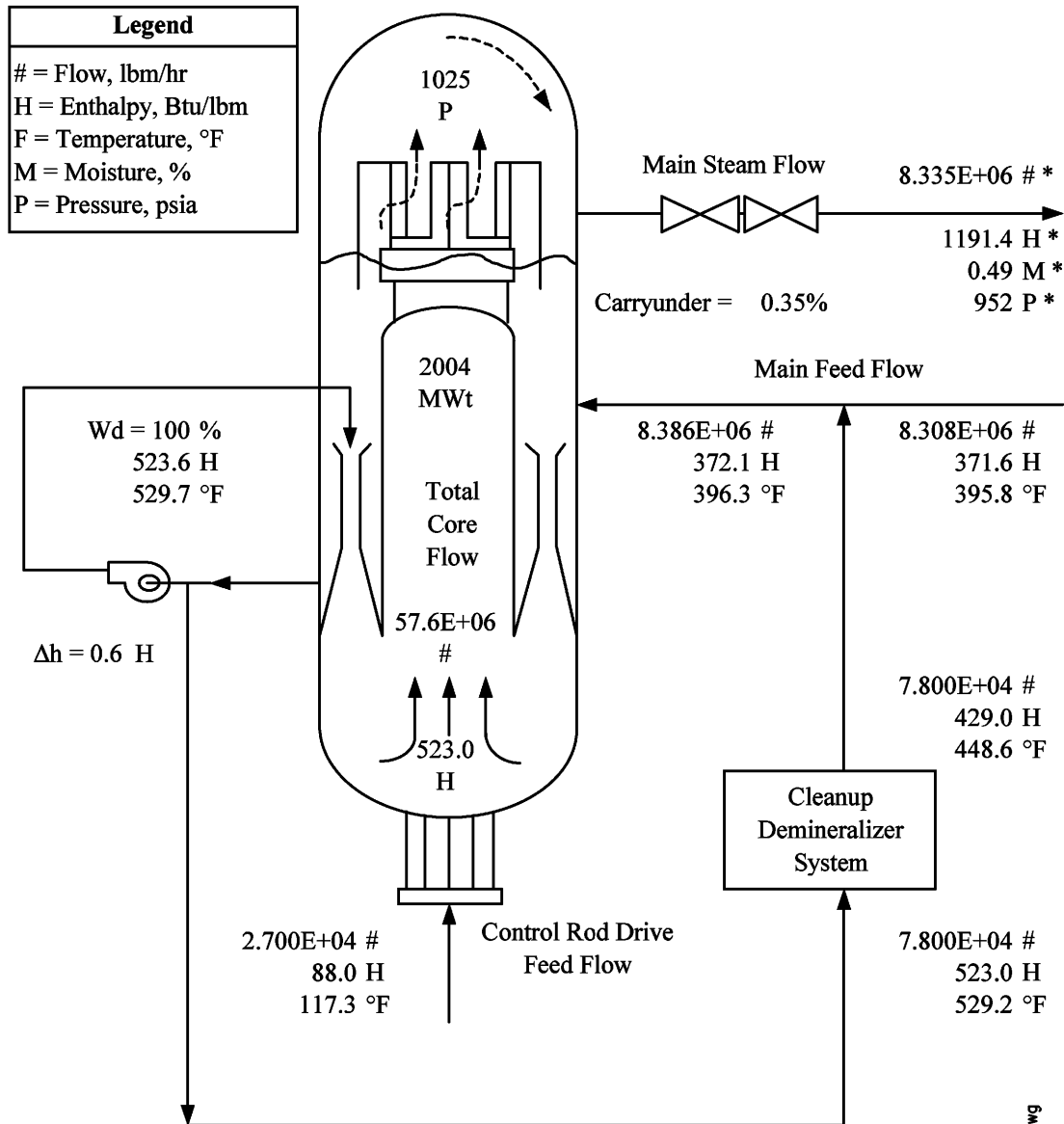
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11. Engineering Change 21225, Rev. 0, "Post RFO25 Feedwater Temperature Impact on EPU Analysis."
12. GE Hitachi Report NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.
13. NRC (T A Beltz) letter to NSPM (K D Fili), "Monticello Nuclear Generating Plant - Issuance of Amendment No. 180 to Renewed Facility Operating License Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. ME3145)," dated March 28, 2014.
14. NSPM letter L-MT-15-074 (Peter A. Gardner) to NRC, "Monticello Extended Power Uprate: Submittal of Power Ascension Testing Results for Replacement Steam Dryer (TAC No. MF6730)," dated October 12, 2015.
15. NRC (T. Beltz) letter to NSPM (K Fili), "Monticello Nuclear Generating Plant - Issuance of Amendment to Revise the Technical Specification to Support Fuel Storage System Changes (TAC No. ME9893) (Am 182), October 24, 2014.
16. NRC (T. Beltz) letter to NSPM (P. Gardner), "Monticello Nuclear Generating Plant - Issuance of Amendment to Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methods (TAC No. MF2479) (Am. 188), June 5, 2015.
17. NRC (R. Kuntz) letter to NSPM (P. Gardner), "Monticello Nuclear Generating Plant- Issuance of Amendment RE: Extended Flow Window (CAC NO. MF5002)," February 23, 2017.

FIGURES

SECTION 1 INTRODUCTION AND SUMMARY

Figure 1.3-1a Normal Reactor Heat Balance 2004 MWt



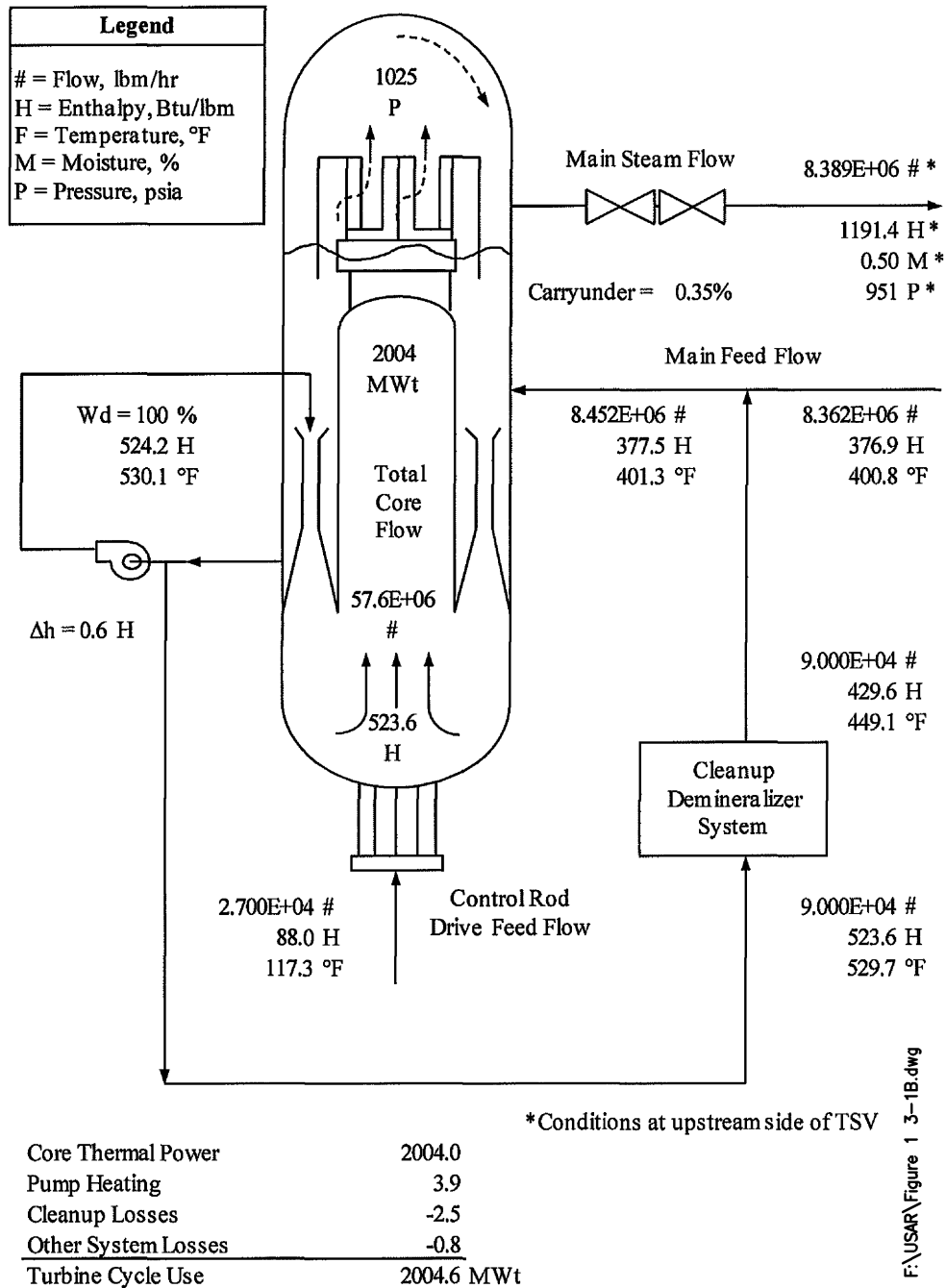
*Conditions at upstream side of TSV

Core Thermal Power	2004.0
Pump Heating	3.9
Cleanup Losses	-2.1
Other System Losses	-0.8
Turbine Cycle Use	2005.0 MWt

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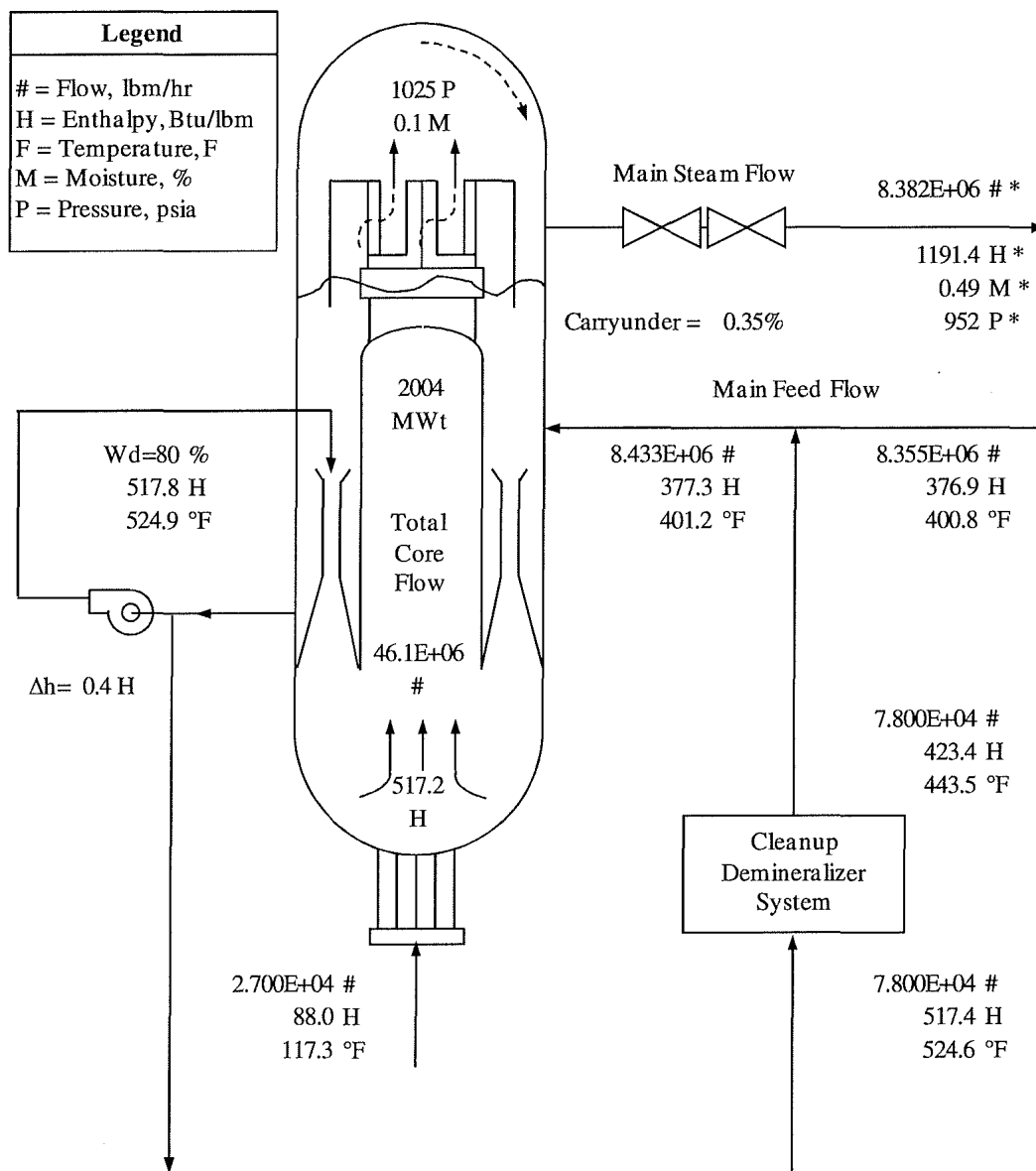
SECTION 1 INTRODUCTION AND SUMMARY

Figure 1.3-1b Increased Feedwater Temperature Reactor Heat Balance 2004 MWt



SECTION 1 INTRODUCTION AND SUMMARY

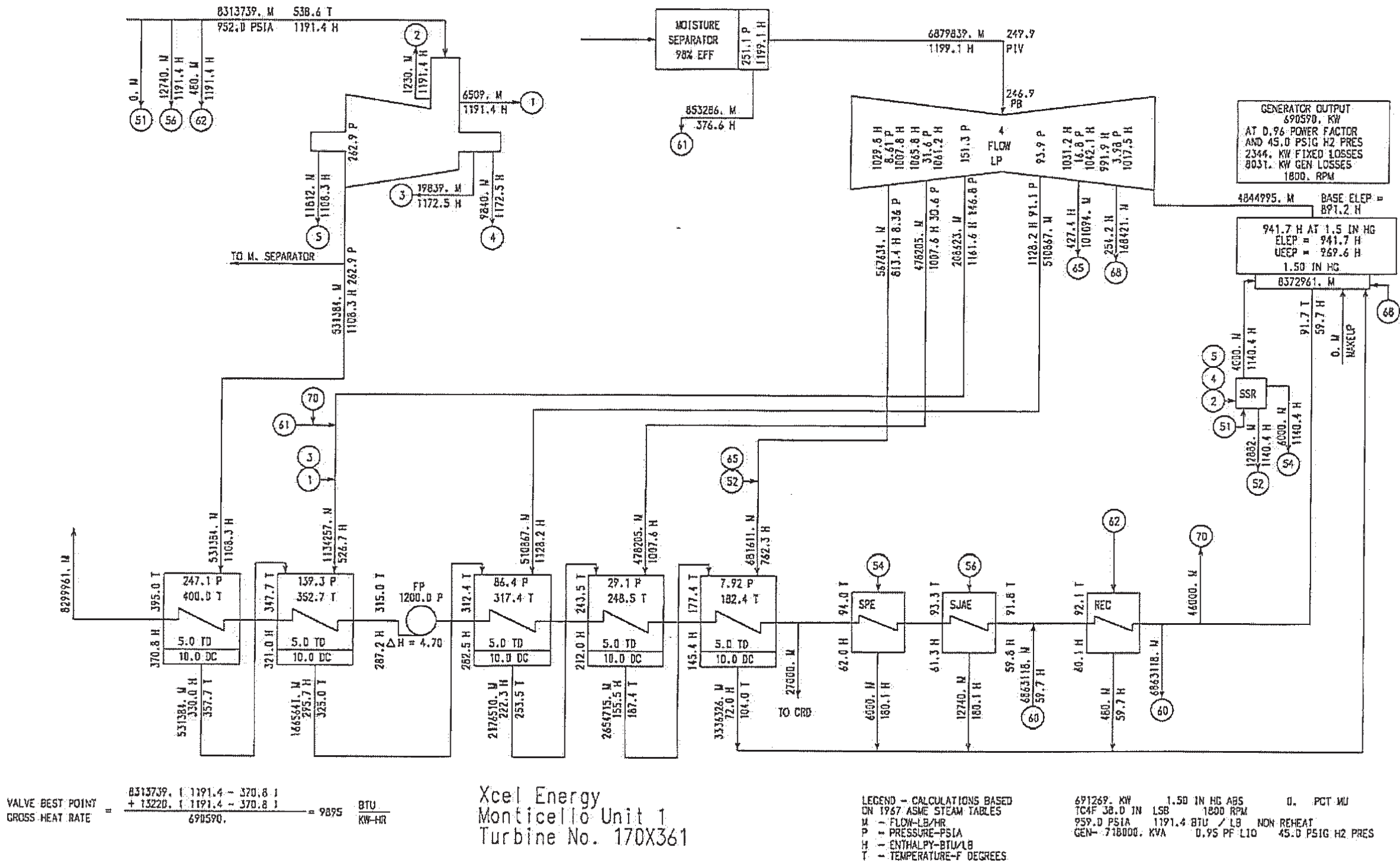
Figure 1.3-1c Increased Feedwater Temperature Reactor Heat Balance 2004 MWt for MELLLA+/EFW



*Conditions at upstream side of TSV

Core Thermal Power	2004.0
Pump Heating	2.0
Cleanup Losses	-2.1
Other System Losses	-0.8
Turbine Cycle Use	2003.1 MWt

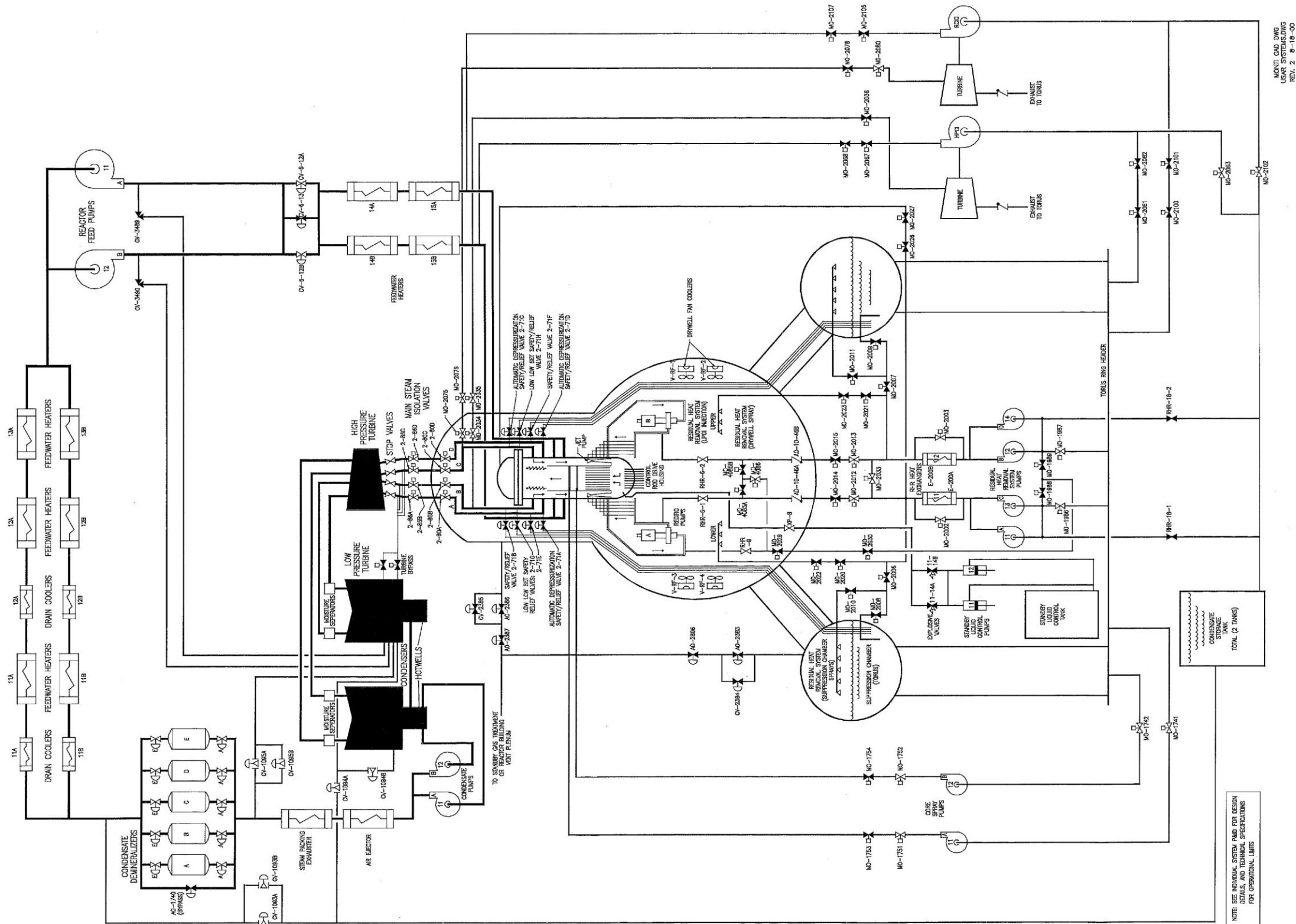
Figure 1.3-2 Turbine Generator System – Heat Balance



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PRIMARY SYSTEM AND EMERGENCY CORE AND CONTAINMENT COOLING SYSTEM

Figure 1.3-3 Plant Process Flow Diagram



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