Nuclear Steam Supply Systems

PERFORMANCE TEST CODES

THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
United Engineering Center
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FOREWORD

The ASME Performance Test Codes Committee voted in June 1965 to establish a test code committee for Nuclear Steam Supply Systems (PTC 32), and later approved as PTC 32 Committee's objective the development of two test codes, PTC 32.1 for Nuclear Steam Supply Systems, and PTC 32.2 for Nuclear Reactor Fuel. These codes are limited in scope to Pressurized Water and Boiling Water Reactors. These types are used in the majority of the nuclear steam supply systems now in operation or under construction in the United States. When other reactor types come into general use, the codes will be expanded or modified to include them.

The PTC 32.1 Code provides a method for testing a nuclear steam supply system as an entity. Published ASME Performance Test Codes are available for the testing of certain components of nuclear steam supply systems (e.g., PTC 8 for Centrifugal Pumps), and it is assumed that such codes will be used, where applicable, when testing of components is required.

A draft of this Code was distributed in March 1968 for comment and criticism by industry and other interested individuals. The Code in final form was approved by the Performance Test Codes Committee on December 3, 1968, and approved and adopted by the Council of the Society by action of the Policy Board, Codes and Standards, on May 19, 1969.

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ASME PERFORMANCE TEST CODES
Test Code for
NUCLEAR STEAM SUPPLY SYSTEMS

CONTENTS

SECTION 0, INTRODUCTION

0.1 This Code contains instructions for the performance testing of nuclear steam supply systems. These systems are defined as nuclear reactors that have been designed for the generation of thermal energy, together with the steam generating and other equipment required to transfer energy from the nuclear fuel to the working fluid of the power cycle.

0.2 This Code is limited to the testing of light water systems using pressurized water reactors (PWR) and boiling water reactors (BWR). This Code will be supplemented as other types of reactors come into general use.

0.3 It is intended that in using this Code a detailed examination will be made of the Code on General Instructions, PTC 1, and all other codes herein referred to, before starting preparations for the tests. Such an examination is for the purpose of assuring an orderly and thorough testing procedure, since it provides the user with an over-all understanding of the ASME Performance Test Codes requirements and enables him to understand readily the interrelationship of the various codes. The latest revisions of the individual codes should be used.

0.4 While Section 2 of this Code is concerned with symbols and their descriptions that apply specifically to the testing of nuclear steam supply systems, the user is referred to the Code of Definitions and Values, PTC 2, for a more complete discussion of the items that will be encountered.

0.5 The Supplements on Instruments and Apparatus, PTC 19, referred to herein, should be applied because the value of the test results depends on the selection and calibration of the instruments and the accuracy of the readings.

0.6 Advanced instrument systems, such as those using electronic devices or mass flow measurement techniques, may, by mutual agreement, be used as alternates to the Code instrument requirements, provided that the application of such instruments has demonstrated accuracy equivalent to that required by this Code.

SECTION 1, OBJECT AND SCOPE

1.01 The purpose of this Code is to establish procedures for conducting tests to determine the thermal performance of a nuclear steam supply system as a unit. These tests include capacity, reactor power level, efficiency, and other related operating characteristics such as steam pressure, moisture content, and solids in the steam. (See Section 4.)

1.02 A determination of any or all of the performance items specified above may be necessary for purposes such as:

1.02.1 Checking the actual performance against guarantee.
1.02.2 Comparing these items with some standard of performance.
1.02.3 Comparing performance at different operating conditions.
1.02.4 Checking performance after new reactor fuel is loaded.
1.02.5 Determining the effects of changes to equipment.
1.02.6 Periodic verification of the means employed to monitor reactor power level.

1.03 The rules and instructions given in this Code apply to the equipment defined in the introduction. Testing of auxiliary apparatus shall be governed by the Performance Test Code applying specifically to the auxiliary apparatus in question.

1.04 Since there is no accurate, direct method of measuring the energy released by the nuclear fuel, the only method used for calculating the energy balance is the direct measurement of the energy output, losses and credits. This is referred to hereinafter as the energy loss method.

1.04.1 The output is defined as the energy absorbed by the working fluid or fluids within the envelope boundary. The working fluid is the steam and water substance through which means the nuclear steam supply system delivers energy.

1.04.2 The input is defined as the energy released by the nuclear fuel plus energy credits added to the working fluid or fluids, air, gas, and other fluid circuits that cross the envelope boundary, as shown in Figure 1 for pressurized water reactors (PWR) and in Figure 3 for boiling water reactors (BWR). The envelope boundary encompasses the equipment to be included in the designation "nuclear steam supply system." The input is equal to the output plus losses.

1.04.3 Energy credits are defined as those amounts of energy added to the envelope of the nuclear steam supply system other than the nuclear energy released by the fuel. These credits include quantities such as pumping energy.

1.04.4 Losses are defined as energy lost from the envelope of the nuclear steam supply system.

1.04.5 For a better understanding of the relationships between input, output, credits, and losses, refer to Figures 2 (PWR) or 4 (BWR).

1.05 The capacity of a nuclear steam supply system is defined as the actual evaporation rate of steam in pounds per hour delivered at specified conditions of the working fluid.

1.06 The reactor power level is defined as the rate of energy release in the reactor core in units of megawatts.

1.07 The efficiency of a nuclear steam supply system, as determined within the scope of this Code, is defined as the ratio of energy absorbed by the working fluid or fluids to energy input as defined in Par. 1.04.2. This definition disregards the energy supplied to the auxiliary apparatus external to the envelope. (See Figures 1 and 3.)

Efficiency is expressed by the following equation:

$$\text{Efficiency (per cent)} = 100 \times \frac{\text{Output}}{\text{Output} + \text{Losses}}$$

1.08 The determination of data of a research nature or other special data is not covered by this Code.

1.09 It is recommended that a report be prepared for each test, giving complete details of the conditions under which the test has been made, including a record of test procedures and all data in a form suitable for demonstrating that the objectives of the test have been attained.

SECTION 2, SYMBOLS AND THEIR DESCRIPTIONS

2.01

SYMBOL	DESCRIPTION	UNIT
A	Surface area of containment	ft^2
B_b	Energy (rate) in blowdown	Btu/hr
B_c	Energy (rate) credits to steam supply system	Btu/hr
B_{ci}	Energy (rate) in reactor inlet cleanup flow	Btu/hr
B_{co}	Energy (rate) in reactor outlet cleanup flow	Btu/hr
B_{cr}	Energy (rate) in control rod drive water	Btu/hr
B_{cs}	Energy (rate) loss in the reactor cleanup system	Btu/hr
B_{cw}	Energy (rate) to cooling water	Btu/hr
B_f	Energy (rate) in feedwater	Btu/hr
B_l	Energy (rate) loss from steam supply system	Btu/hr
B_{ld}	Energy (rate) in letdown from reactor coolant system	Btu/hr
B_m	Energy (rate) in makeup to reactor coolant system	Btu/hr
B_{rl}	Energy (rate) loss from energy envelope by radiation or convection	Btu/hr
B_s	Energy (rate) in steam	Btu/hr
B_{sw}	Energy (rate) in seal water	Btu/hr
B_w	Energy (rate) absorbed by working fluid	Btu/hr
C	System energy capacity	Mw hr/F
h_b	Enthalpy of blowdown	Btu/lb
h_{cr}	Enthalpy of control rod drive water	Btu/lb
h_f	Enthalpy of feedwater	Btu/lb
h_i	Enthalpy of cooling water to cooling system	Btu/lb
h_{ld}	Enthalpy of reactor coolant letdown	Btu/lb
h_m	Enthalpy of reactor coolant makeup	Btu/lb
h_o	Enthalpy of cooling water from cooling system	Btu/lb
h_s	Enthalpy of steam at steam supply system outlet	Btu/lb
h_{swi}	Enthalpy of seal water inlet	Btu/lb
η	Nuclear steam supply system efficiency	%
P_{me}	Average electrical power input to miscellaneous electrical equipment inside energy envelope	kw
P_{pe}	Average electrical power input to reactor coolant circulating pump driver	kw
P_{pre}	Average electrical power input to pressurizer heaters	kw
P_r	Reactor power level	Mw
P_s	Pressure of steam at steam supply system outlet	psia
t_f	Feedwater temperature	F
t_i	Containment (ambient) temperature	F
t_o	Outside air temperature	F
t_s	Steam temperature at steam supply system outlet	F
t_{swi}	Temperature of inlet seal water	F
U	Calculated heat transfer coefficient	Btu/hr-ft²-F
W_b	Blowdown flow	lb/hr
W_{cr}	Water flow to control rod drives	lb/hr
W_{cs}	Cleanup system cooling water flow	lb/hr
W_{cw}	Cooling water flow	lb/hr
W_f	Feedwater flow	lb/hr
W_{ld}	Reactor coolant letdown flow	lb/hr
W_m	Reactor coolant makeup flow	lb/hr
W_s	Steam flow from nuclear steam supply system	lb/hr
W_{swi}	Water flow to seals	lb/hr
W_{swo}	Water flow from seals	lb/hr

x	Steam quality at steam supply system outlet	mass per cent
Δh_{cs}	Change in enthalpy in cleanup system cooling water	Btu/lb
Δt	Time rate of change of reactor coolant temperature	F/hr

2.02 Test and Run. Throughout this Code the word "test" is applied only to the entire investigation, and the word "run" to a subdivision. A run consists of a complete set of observations made for a period of time with one or more of the independent variables maintained virtually constant.

SECTION 3, GUIDING PRINCIPLES

3.01 Items on Which Agreement Shall Be Reached. In order to achieve the objectives of the test, the parties to the test must reach agreement on the following pertinent items:

3.01.1 Envelope Boundary. This Code is based on the use of the loss method for determining the energy balance. Energy credits and losses are determined with reference to an envelope boundary, such as shown in Figures 1 and 3 for the PWR and BWR, respectively. The space within the envelope boundary represents the region of energy release within the nuclear steam supply system to which certain credits are added and from which certain losses are subtracted. The envelope boundary need not necessarily represent any particular physical boundary, such as the containment vessel or the reactor coolant system. The definition of the envelope boundary shall be established by mutual agreement to include or exclude certain pieces of equipment where this is appropriate.

3.01.2 Capacity determination – defined in Par. 1.05.

3.01.3 Energy credits to be measured.

3.01.4 Energy credits to be assigned where not measured.

3.01.5 Energy losses to be measured.

3.01.6 Energy losses to be assigned where not measured.

3.01.7 Permissible deviation from specified operating conditions between duplicate runs.

3.01.8 Power level determination – defined in Par. 1.06.

3.01.9 Efficiency determination – defined in Par. 1.07.

3.01.10 Other related operating characteristics – Pars. 4.02 and 4.03.

3.01.11 Allocation of responsibility for all performance and operating conditions which affect the test.

3.01.12 Selection of test personnel to conduct the test.

3.01.13 Establishment of acceptable operational conditions such as control rod positions, fluid temperatures and pressures, water level in steam drums, reference system arrangement, chemical additions to incoming feed, leakage rates, etc.

3.01.14 Cleanliness of the system initially and during the test.

3.01.15 The source of thermodynamic properties to be used. The latest edition of the ASME Steam Tables is recommended.

3.01.16 Establishment of number and duration of runs, number of load points, observations and readings to be taken to comply with the object or objectives of the test, procedures to be followed, and basis for rejection of runs.

3.01.17 Instruments, calibration of instruments, methods of measurement, and equipment to be used in testing the unit. The Performance Test Code Supplements on Instruments and Apparatus should be used when applicable.

3.01.18 Corrections to be made for deviations from specified operating conditions.

3.02 Selection of Personnel. To insure obtaining reliable results, all personnel participating in the test shall be fully qualified to perform their particular function.

3.03 Tolerances and Limits of Error. This Code does not include consideration of over-all tolerances or margins on performance guarantees. The test results shall be reported as computed from test observations, with proper corrections for calibrations.

3.03.1 Allowances for errors of measurement and sampling are permissible provided they are agreed upon in advance by the parties to the test and clearly stated in the test report.

3.03.2. Whenever allowances for probable errors of measurement and sampling are to be taken into consideration, the reported test results shall be qualified by the statement that the error in the results may be considered not to exceed a given plus or minus percentage.

3.04 Acceptance Test. An acceptance test shall be undertaken only when the parties to the test certify that the unit is operational and ready for test.

3.04.1 Parties to the test should designate a person to direct the test, and may designate an arbiter in the event of disputes as to the accuracy of observations and conditions or methods of operation.

3.04.2 After a preliminary run has been made, it may be declared an acceptable run if agreed to, provided that all the requirements of a regular run have been met.

3.04.3 At least two runs shall be made to demonstrate the attainment of capacity or reactor power level. If the results exceed the previously agreed upon deviation between runs, then the conditions of each test should be analyzed to determine whether there is reason to reject either or both tests by mutual agreement. The permissible deviation should also be re-evaluated to determine whether there is reason to change the permissible deviation by mutual agreement. If this examination does not result in mutual agreement that the two tests are acceptable and the deviation is acceptable, additional runs shall be made until at least two runs fall within the agreed permissible deviation. Acceptance shall be based upon the average of these two runs.

3.05 Preparation for All Tests

3.05.1 The entire system shall be checked for leakage and, where necessary, appropriate corrective actions shall be undertaken.

3.05.2 Any departures from standard or previously specified conditions in the physical state of equipment, cleanliness, or operating conditions shall be corrected if practicable. Otherwise, such departures shall be described clearly in the report of the test.

3.06 A preliminary run shall be made for the purpose of:

3.06.1 Determining that the nuclear steam supply system is in a suitable condition for conducting the test.

3.06.2 Checking operation of all instruments and test apparatus.

3.06.3 Training the test personnel.

3.07 Constancy of Test Conditions

3.07.1 Preparatory to any test run, the nuclear steam supply system shall be operated for a sufficient time to show that steady operating conditions have been attained. The parties to the test shall agree as to when these conditions have been attained.

3.07.2 Special means, as appropriate, may be employed for securing constant load.

3.07.3 Feedwater flow, water levels, and all controllable temperatures and pressures shall remain, as nearly as possible, unchanged throughout the run. Any other conditions in which variations might significantly affect the results of the test, shall also be held as constant as possible throughout the test.

3.07.4 Every effort shall be made to conduct tests under the specified conditions such as output, pressures, and temperatures, or as close to the specified conditions as possible in order to minimize the application of corrections to the test results. For each variable, the parties to the test shall agree to the maximum permissible deviation of the average test value from the specified condition.

3.07.5 Should serious inconsistencies in the observed data be detected, the run shall be rejected completely, or in part if the affected part is at the beginning or end of the run.

3.08 Duration of Runs

3.08.1 When determining the energy balance, each run should be preferably of not less than four hours duration.

3.08.2 The actual duration of all runs from which the final data are derived shall be clearly stated in the test report.

3.09 Frequency and Consistency of Readings. Except for quantity measurements, it is recommended that the readings be taken at 10-minute intervals. If, however, there are significant fluctuations, the readings shall be taken at such frequency as may be necessary to determine a representative average. In this case, the parties to the test shall agree to the reading frequency to be used.

3.09.1 Where the amount of feedwater is determined from integrating instruments, a reading shall be taken every hour. When indicating flowmeters or manometers are used with venturi tubes, flow nozzles, flow tubes, or orifice plates for subsequently determining quantity measurements, the flow indicating element shall be read at intervals not exceeding five minutes. Readings shall be taken at intervals as short as one minute when deemed necessary by the parties to the test.

3.09.2 It is suggested that, insofar as feasible, pertinent data of the run be plotted continuously, as the run progresses, on coordinate paper of suitable scale arrangements to permit a complete review of the conduct of the run at least hourly.

3.10 Records and Test Reports. All observations, measurements, and instrument readings necessary for the objective of the test shall be recorded as observed. Corrections and corrected values shall be entered separately in the test record. Erasures in this record shall be prohibited. If deviations from standard or previously specified conditions are accepted by parties to the tests, they should be described in the test report.

3.11 Instruments and Methods of Measurement. The necessary instruments and procedures for making measurements are prescribed herein and should be used in conjunction with the following ASME Performance Test Codes Supplements on Instruments and Apparatus, and other publications for detailed specifications on apparatus and procedures involved in the testing. In all cases, care shall be exercised to apply the latest revision of the document concerned.

3.11.1 ASME Performance Test Codes:

General Instructions, PTC 1
Definitions and Values, PTC 2
Steam Generating Units, PTC 4.1
Steam Turbines, PTC 6

3.11.2 Supplements on Instruments and Apparatus, PTC 19:

Part 1 – General Considerations, PTC 19.1
Part 2 – Pressure Measurement, PTC 19.2
Part 3 – Temperature Measurement, PTC 19.3
Part 5 – Measurement of Quantity of Materials, PTC 19.5
Part 6 – Electrical Measurements in Power Circuits, PTC 19.6
Part 11 – Methods for Determination of Quality and Purity of Steam, PTC 19.11
Part 21 – Leak Detection and Leakage Measurement, PTC 19.21

3.11.3 ASME Research Publications:
Fluid Meters: Their Theory and Application

3.11.4 ASTM Standards, Part 23:
D-1066 – Method of Sampling Steam
D-1125 – Methods of Test for Electrical Conductivity of Industrial Water and Industrial Waste Water
D-1428 – Methods of Test for Sodium and Potassium in Industrial Water and Water-Formed Deposits by Flame Photometry
D-2186 – Methods of Test for Deposit-Forming Impurities in Steam

SECTION 4, SYSTEM THERMAL PERFORMANCE

4.01 Determination of Capacity, Reactor Power Level and Efficiency. The capacity of a nuclear steam supply system is defined as the rate of steam generation in pounds per hour delivered at specified conditions of the working fluid. The power level of the reactor is equal to the rate of fuel energy input and is usually expressed in megawatts rather than in Btu per hour. The loss method, by which the energy balance is calculated, shall be based on accurate and complete information which will make possible the determination of output, losses, and credits. The efficiency is equal to the output divided by the sum of the output and losses.

4.02 Data Required

4.02.1 Accurate data on the following items are required:

4.02.1.1 Temperature, pressure, and flow rates of any medium representing energy quantities in the output or losses, and also the power inputs to other items which represent credits, as indicated in Figures 2 and 4.

4.02.1.2 Quality of steam where appropriate.

4.02.1.3 Pump power.

4.02.2 Certain items may be determined to be insignificant in comparison with the limits of accuracy of the power level determination and may be disregarded by agreement.

4.02.3 Certain other items may be found to be small, yet significant enough to include in the efficiency and power level determinations. In these cases, it may be determined that a correlation can be made with reactor power level for these items and that subsequent determinations of reactor power can utilize these test results without use of further new data.

4.02.4 Where access for readings may not be permissible due to the presence of nuclear radiation, or where instrument connections may require excessive length, calibrated transmitters may be used.

4.03 Measurements

4.03.1 Flow Measurement of Feedwater. The output steam flow rate to be used in the determination of capacity, reactor power level and efficiency shall be obtained from feedwater flow measurement. This flow rate must be corrected for any addition or withdrawal of fluid, such as blowdown or injection water, downstream of the measuring element.

Water quantity may be measured by venturi tube, flow nozzle, flow tube, or thin plate orifice. These measuring devices shall be calibrated prior to the test and corrected to measure the full-capacity flow within ± 0.50 per cent. Read-out instrumentation, plus measuring devices, shall measure this flow within ± 0.70 per cent.

4.03.1.1 The recommendations of ASME Fluid Meters,* Part II, Flow Measurement, shall be consulted with reference not only to the design, construction, calibration, and use of flow measuring elements, but also to their location and installation in the pipe lines and the installation of the connecting piping system between the primary element and the measuring device. All computations of flow rate from the observed differentials, pressures, and temperatures shall be made in accordance with the provisions of ASME Fluid Meters,* Part II.

*ASME Fluid Meters, Part II, has not yet been issued. Pending its issue, expected in 1970, PTC 19.5;4 and PTC 6 (1964) should be consulted. Additional information is also available in a supplementary report prepared by the PTC 6 Committee entitled "Report on Guidance for Evaluation of Performance Tests of Steam Turbines."

within a pipe, the temperature should be measured downstream of one or preferably two elbows, but not to exceed 50 ft from the normal measuring point.

4.03.4.2 All temperature measuring instruments and wells shall be constructed, installed, calibrated before and after test, and operated in accordance with I and A, Temperature Measurement, PTC 19.3.

4.03.5 Measurement of Pressures. Pressures shall be measured within ± 0.5 per cent. The instrument recommended for pressure measurement above 35 psia or beyond mercury manometer range, is the calibrated deadweight gage. Calibrated laboratory Bourdon gages may be used to determine pressure where a high degree of accuracy is not required.

The following precautions shall be observed in the use of pressure measuring devices:

4.03.5.1 Where applicable all pressure measuring instruments shall be constructed and installed, in accordance with I and A Pressure Measurement, PTC 19.2, and shall be calibrated before and after test.

4.03.5.2 Pressure gages shall be located where they will not be affected by any disturbing influences such as extremely high or low temperatures or vibration and shall be located in convenient positions for reading.

4.03.5.3 Gage connections shall be as short and direct as possible and shall be free from leaks.

4.03.5.4 Pressure gage pulsations shall not be dampened by throttling the connection to the gage or by the use of commercial gage dampers, but a volume chamber may be employed. The arrangement may be considered satisfactory if the maximum and minimum values of the instantaneous pressures do not differ by more than 2.0 per cent from the mean value.

4.03.6 Moisture and Solids in Steam. The following methods may be used to determine steam quality and purity:

(1) Flame photometry method for sodium
(2) Radioactive tracer method
(3) Electrical conductivity method for dissolved solids
(4) Gravimetric method for total solids
(5) Calorimeter for direct determination of quality
(6) Additional method if mutually agreed upon.

Selection of one of these methods for determining steam quality or purity must be based upon the conditions peculiar to a particular steam supply system, since each method has limitations which govern its use.

4.03.6.1 Flame Photometry Method for Sodium. The sodium flame photometry method is based upon the accurate measurement of sodium concentrations in condensed steam samples and is therefore contingent upon the presence of sodium salts in the boiler water of the steam supply system. It is not recommended where a large percentage of the solids present do not contain sodium. Sodium analysis shall be performed in accordance with ASTM D-1428, Method B, and ASTM D-2186, Method C.

Solids in the steam may be calculated as follows:

Total Solids in Steam (ppm) =

$$\frac{\text{Sodium in Steam (ppm)}}{\text{Sodium in Boiler Water (ppm)}} \times \begin{matrix}\text{Total Solids in} \\ \text{Boiler Water (ppm)}\end{matrix}$$

Steam quality may be calculated as follows:

$$x\ (\%) = 100 - \left[\frac{\text{Sodium in Steam (ppm)} \times 100}{\text{Sodium in Boiler Water (ppm)}}\right]$$

The accuracy of this method for determining quality is impaired when volatile salts are present in the steam.

4.03.6.2 Radioactive Tracer Method. This method is based upon the accurate measurement of radioactive nuclides, such as sodium-24 in condensed steam and boiler water samples. The activity of the samples can be determined by using a multichannel analyzer with the energy spectrum and half-life used to identify the specific nuclide.

Steam quality can be calculated as follows:

$$x\ (\%) = 100 - \left[\frac{\text{Activity in Steam} \times 100}{\text{Activity in Boiler Water}}\right]$$

This method is particularly applicable to boiling water reactors where radioactive isotopes are normally present in both the boiling water and steam effluent. The relative activities of the samples must permit good counting statistics. A specific PTC or ASTM reference is not presently available for this method.

4.03.6.3 Electrical Conductivity Method for Dissolved Solids. When the total solids in the steam are in excess of 0.5 ppm, the electrical conductivity method may be used to determine steam purity and steam quality. This method requires the presence of significant dissolved salt concentrations in the boiler water of the steam supply system. The determination of dissolved solids in condensed steam samples shall be made in accordance with PTC 19.11 or ASTM D-1125 and D-2186, Method B. For measuring performance by the electrical conductivity method, the average of ten determinations made at regular intervals throughout the test period shall be used.

Steam quality may be calculated as follows:

$$x\,(\%) = 100 - \left[\frac{\text{Dissolved Solids in Steam}}{\text{Dissolved Solids in Boiler Water}} \times 100\right]$$

In addition to the interference of dissolved gases, the presence of volatile salts in the steam sample impairs the accuracy of this method.

4.03.6.4 Gravimetric Method for Total Solids. Steam purity and quality may be determined by measuring the total solids in condensed steam samples by the gravimetric method outlined in PTC 19.11, Section 3. For measuring performance by the gravimetric method, the results shall be expressed as the average of three determinations made upon a composite sample, which shall be taken throughout the entire test period.

Steam quality may be calculated as follows:

$$x\,(\%) = 100 - \left[\frac{\text{Total Solids in Steam}}{\text{Total Solids in Boiler Water}} \times 100\right]$$

4.03.6.5 Calorimeter for Direct Determination of Quality. When the steam supply system boiler water is free of solids, steam quality must be determined by the use of a suitable calorimeter as defined in PTC 19.11, Section 2. Calorimeters must be installed at the point of sampling; hence, in nuclear steam supply systems radiation hazards may require that remote reading instrumentation be used instead of direct reading instruments. The potential for cumulative errors in such an installation should be reviewed and agreed to by the parties of the test before the tests are run.

4.03.6.6 Additional Method. An additional method, known as the "Sodium Ion Electrode Method," is now being used for the determination of steam quality and purity. This method employs an electrode and instrument similar to those used for the measurement of the hydrogen-ion concentration in aqueous solutions (pH), except that the electrode is specific for measuring the sodium ion concentration. This device is a potential alternative to the flame spectrophotometer as described in Section 4.03.6.1.

The use of this device is a new technique that has not had sufficient operating experience to provide a PTC or ASTM procedure for reference.

4.03.6.7 Sampling Techniques

4.03.6.7.1 Steam Sampling. Steam sampling, with respect to the type of probe and its location, and method of withdrawing the sample shall be performed in accordance with the ASTM "Method of Sampling Steam, D-1066." Inherent limitations in the probe sample technique make it extremely difficult to obtain a representative steam line sample.

4.03.6.7.2 Hot Well, Condensate, or Feedwater Sampling. This method uses a condensed sample from the feedwater or condensate systems to represent the steam generator output instead of a sampling nozzle. The ratio of sodium conductivity in this sample to a boiler water sample can be measured by one of the previously mentioned analyses. The concentrations of the representative steam sample and the boiler water sample are to be corrected for background sodium by subtracting base load values from values obtained at load.

The potential for salt deposition on system components exposed to the steam cycle is a matter of concern when using this method. The subsequent escape of sodium back into the system can contribute to variable or indeterminate readings. Any condenser leakage or external contamination will affect the accuracy of this method, especially when the magnitude of these factors is unknown. The acceptable limits of error and operational difficulties associated with this method must be clearly defined and understood by all parties before the method is considered for performance testing.

4.03.7 Electrical Measurements. Several energy inputs are in the form of electrical energy, and some of this electrical energy appears as an input to the fluid stream it serves.

4.03.7.1 Alternating current power shall be determined in accordance with PTC 19.6, Section 5, using a calibrated instrument suitable for auxiliary power measurement.

SECTION 5, COMPUTATIONS

5.01 Pressurized Water Reactor. The computations in paragraphs 5.02 to 5.04 are to be used for calculating capacity, reactor power level, and nuclear steam supply system efficiency of a pressurized water reactor system. The calculations are intended to represent a typical pressurized water system, and variations in the calculations for any system should be established and agreed to by the parties to the test.

5.02 Capacity, W_s, as defined in Par. 1.05.

$W_s = W_f - W_b$

where:

W_s = steam flow from nuclear steam supply system, lb/hr

W_f = feedwater flow, lb/hr

W_b = blowdown flow, lb/hr.

Specified Conditions:

P_s = pressure of steam at steam supply system outlet, psia

t_s = steam temperature at steam supply system outlet, F

x = steam quality at steam supply system outlet, mass per cent

t_f = feedwater temperature, F.

5.03 Reactor power level, P_r, as defined in Par. 1.06.

$$P_r = \frac{B_w + B_l - B_c}{3{,}412{,}141}$$

where:

P_r = reactor power level, megawatts

B_w = energy (rate) absorbed by working fluid, Btu/hr

B_l = energy (rate) losses from steam supply system, Btu/hr

B_c = energy (rate) credits to steam supply system, Btu/hr.

5.03.1 Energy (rate) absorbed by working fluid, B_w.

$B_w = B_s + B_b - B_f$

where:

B_s = energy (rate) in steam, Btu/hr

B_b = energy (rate) in blowdown, Btu/hr

B_f = energy (rate) in feedwater, Btu/hr.

Note: Energy absorbed by the working fluid must be calculated for each steam generator, and the *total* absorbed energy shall be B_w.

5.03.1.1 Energy (rate) in steam, B_s.

$B_s = (W_f - W_b)\ h_s$

where:

h_s = enthalpy of steam, at steam supply system outlet, Btu/hr.

5.03.1.2 Energy (rate) in blowdown, B_b.

$B_b = W_b\ h_b$

where:

h_b = enthalpy of blowdown, Btu/lb.

5.03.1.3 Energy (rate) in feedwater, B_f.

$B_f = W_f\ h_f$

where:

h_f = enthalpy of feedwater, Btu/lb.

5.03.2 Energy (rate) loss, B_l.

$B_l = B_{ld} - B_m - B_{sw} + B_{cw} + B_{rl}$

where:

B_{ld} = energy (rate) in letdown from reactor coolant system, Btu/hr

B_m = energy (rate) in makeup to reactor coolant system, Btu/hr

B_{sw} = energy (rate) in seal water, Btu/hr

B_{cw} = energy (rate) to cooling water, Btu/hr

B_{rl} = energy (rate) loss from energy envelope by radiation or convection, Btu/hr.

5.03.2.1 Energy (rate) in letdown from reactor coolant system, B_{ld}.

$B_{ld} = W_{ld}\ h_{ld}$

where:

W_{ld} = reactor coolant letdown flow, lb/hr

h_{ld} = enthalpy of reactor coolant letdown, Btu/lb.

5.03.2.2 Energy (rate) in makeup to reactor coolant system, B_m.

$B_m = W_m\ h_m$

where:

W_m = reactor coolant makeup flow, lb/hr

h_m = enthalpy of reactor coolant makeup, Btu/lb.

5.03.2.3 Energy (rate) in seal water, B_{sw}.

For purposes of computation the temperatures of the seal water entering and leaving the envelope are assumed to be equal.

$B_{sw} = (W_{swi} - W_{swo})\ h_{swi}$

where:

W_{swi} = water flow to seals, lb/hr

W_{swo} = water flow from seals, lb/hr

h_{swi} = enthalpy of seal water inlet, Btu/lb.

5.03.2.4 Energy (rate) to cooling water, B_{cw}.

$B_{cw} = W_{cw}\ (h_o - h_i)$

where:

W_{cw} = cooling water flow, lb/hr

h_o = enthalpy of cooling water from cooling system, Btu/lb

h_i = enthalpy of cooling water to cooling system, Btu/lb.

5.03.2.5 Energy (rate) loss from energy envelope by thermal radiation and convection, B_{rl}.

Thermal radiation and convection losses shall be determined, preferably by testing, and confirmed if possible by calculations. The exact method shall depend on definition of the envelope boundary chosen for the energy balance.

If, for example, reactor coolant system is defined as the envelope, energy losses shall be measured by testing the nuclear steam supply system during a period when no steam is being removed from the system. A measurement of the time rate of change of reactor coolant temperatures in conjunction with calculated values of system energy capacity in Mw-hr/F will determine system energy losses as follows:

$B_{rl} = P_{pe} - C\Delta t$

where:

P_{pe} = Average electrical power input to reactor coolant circulating pump driver, kw (assuming this is the only energy input). (If other factors contribute, such as reactor decay heat, they also must be included.)

C = System energy capacity, Mw-hr/F, determined by calculation of mass times specific heat of coolant system.

Δt = Time rate of change of reactor coolant temperature, F/hr, determined when the system is isolated.

If, for example, the reactor containment defined the envelope, energy losses from the containment could be calculated as follows:

$B_{rl} = UA\ (t_i - t_o)$

where:

U = a calculated heat transfer coefficient, Btu/hr-ft²-F

A = surface area of containment, ft²

t_i = containment ambient temperature, F

t_o = outside air temperature, F

Nonuniform temperature distribution inside and outside of containment, nonuniform containment wall thickness, or nonuniform heat transfer media must be taken into account.

5.03.3 Energy (rate) credits, B_c.

$B_c = (P_{pre} + P_{pe} + P_{me})\ 3412.141$

where:

P_{pre} = average electrical power input to pressurizer heaters, kw

P_{pe} = average electrical power input to reactor coolant circulating pump driver, kw (when the driver is outside the energy envelope, the driver efficiency must be included)

P_{me} = average electrical power input to miscellaneous electrical equipment inside the energy envelope, kw.

5.04 Nuclear steam supply system efficiency (per cent), η, as defined in Par. 1.07.

$$\eta = (B_w)(100) / (B_w + B_l) = \frac{(B_s + B_b - B_f)\ 100}{(B_s + B_b - B_f + B_{ld} - B_m - B_{sw} + B_{cw} + B_{rl})}$$

5.05 Boiling Water Reactor. The computations in paragraphs 5.06 to 5.08 are to be used for calculating capacity, reactor power level, and nuclear steam supply system efficiency of a direct cycle boiling water reactor system. The calculations are intended to represent a typical boiling water system, and variations in the calculations for any system should be established and agreed to by the parties to the test.

5.06 Capacity, W_s, as defined in Par. 1.05.

$W_s = W_f + W_{swi} - W_{swo} + W_{cr} - W_b$

where:

W_s = steam flow from nuclear steam supply system, lb/hr
W_f = feedwater flow, lb/hr
W_{swi} = water flow to seals, lb/hr
W_{swo} = water flow from seals, lb/hr
W_{cr} = water flow to control rod drives, lb/hr
W_b = blowdown flow, lb/hr

Specified conditions:

P_s = pressure of steam at steam supply system outlet, psia
t_s = steam temperature at steam supply system outlet, F
x = steam quality at steam supply system outlet, mass per cent
t_f = feedwater temperature, F
t_{swi} = temperature of inlet seal water, F.

5.07 Reactor power level, P_r, as defined in Par. 1.06.

$$P_r = \frac{B_w + B_l - B_c}{3,412,141}$$

where:

P_r = reactor power level, megawatts
B_w = energy (rate) absorbed by working fluid, Btu/hr
B_l = energy (rate) losses from steam supply system, Btu/hr
B_c = energy (rate) credits to steam supply system, Btu/hr.

5.07.1 Energy (rate) absorbed by working fluid, B_w:

$B_w = B_s - B_f - B_{sw} - B_{cr} + B_b$

where:

B_s = energy (rate) in steam, Btu/hr
B_f = energy (rate) in feedwater, Btu/hr
B_{sw} = energy (rate) in seal water, Btu/hr
B_{cr} = energy (rate) in control rod drive water, Btu/hr
B_b = energy (rate) in blowdown, Btu/hr.

5.07.1.1 Energy (rate) in steam, B_s.

$B_s = (W_f + W_{swi} - W_{swo} + W_{cr} - W_b)\ h_s$

where:

h_s = enthalpy of steam at steam supply system outlet, Btu/lb.

5.07.1.2 Energy (rate) in feedwater, B_f.

$B_f = W_f\ h_f$

where:

h_f = enthalpy of feedwater, Btu/lb.

5.07.1.3 Energy (rate) in seal water, B_{sw}. For purposes of computation the temperatures of the seal water entering and leaving the envelope are assumed to be equal.

$B_{sw} = (W_{swi} - W_{swo})\ h_{swi}$

where:

h_{swi} = enthalpy of seal water inlet, Btu/lb.

5.07.1.4 Energy (rate) in control rod drive water, B_{cr}.

$B_{cr} = W_{cr}\ h_{cr}$

where:

h_{cr} = enthalpy of control rod drive water, Btu/lb.

5.07.1.5 Energy (rate) in blowdown, B_b.

$B_b = W_b h_b$

where:

h_b = enthalpy of blowdown measured at point of removal from reactor water cleanup system, Btu/lb.

5.07.2 Energy (rate) loss, B_l.

$B_l = B_{cw} + B_{rl} + B_{cs}$

where:

B_{cw} = energy (rate) to cooling water, Btu/hr
B_{rl} = energy (rate) loss from energy envelope by radiation or convection, Btu/hr
B_{cs} = energy (rate) loss in the reactor cleanup system, Btu/hr.

5.07.2.1 Energy (rate) loss from cooling water, B_{cw}

$B_{cw} = W_{cw}\ (h_o - h_i)$

where:

W_{cw} = cooling water flow, lb/hr
h_o = enthalpy of cooling water from cooling system, Btu/lb
h_i = enthalpy of cooling water to cooling system, Btu/lb.

5.07.2.2 Energy (rate) loss from energy envelope by thermal radiation and convection, B_{rl}.

Thermal radiation and convection losses shall be determined, preferably by testing, and confirmed if possible by calculations. The exact method shall depend on definition of the envelope boundary chosen for the energy balance.

If, for example, the reactor coolant system is defined as the envelope, energy losses shall be measured by testing the nuclear steam supply system during a period when no steam is being removed from the system. A measurement of the time rate of change of system pressure, for which saturation temperatures may be determined, in conjunction with measured values of system energy capacity in Mw-hr/F will determine system energy losses as follows:

$B_{rl} = C\Delta t$

where:

C = system energy capacity, Mw-hr/F, determined by a measurement of the time rate of change of reactor coolant temperature for a known energy input such as from the reactor circulation pumps.

Δt = time rate of change of reactor coolant temperature, F/hr, determined by a measurement of the time rate of change of nuclear system saturation pressure when the system is isolated.

If, for example, the reactor containment defined the envelope, heat losses from the containment could be calculated as follows:

$B_{rl} = UA\ (t_i - t_o)$

where:

U = a calculated heat transfer coefficient, Btu/hr-ft^2-F

A = surface area of containment, ft^2

t_i = containment building temperature, F

t_o = outside air temperature, F.

Nonuniform temperature distribution inside and outside of containment, nonuniform containment wall thickness, or nonuniform heat transfer media must be taken into account.

5.07.2.3 Energy (rate) loss in reactor cleanup system, B_{cs}.

$B_{cs} = W_{cs}\Delta h_{cs}$

where:

W_{cs} = cleanup system cooling water flow, lb/hr

Δh_{cs} = change in enthalpy in cleanup system cooling water, Btu/lb.

5.07.3 Energy (rate) credits, B_c.

$B_c = (P_{pe} + P_{me})\ 3412.141$

where:

P_{pe} = average electrical power input to reactor coolant recirculating pump driver, kw (when the driver is outside the energy envelope the driver efficiency must be included).

P_{me} = average electrical power input to miscellaneous electrical equipment inside the energy envelope, kw.

5.08 Nuclear steam supply system efficiency (per cent), η, as defined in Par. 1.07.

$$\eta = \frac{B_w}{(B_w + B_l)} \times 100 =$$

$$\frac{(B_s - B_f - B_{sw} - B_{cr} + B_b)\ 100}{(B_s - B_f - B_{sw} - B_{cr} + B_b + B_{cw} + B_{rl} + B_{cs})}$$

SECTION 6, REPORTING RESULTS

6.01 The ASME test forms for boiling water and pressurized water reactors include the relevant items for a typical nuclear steam supply system; however, each form should be supplemented or changed to suit the characteristics of the particular system being tested. The parties to the test should agree to those parameters to be measured in accordance with Par. 3.01.

TEST FORMS

AND

FIGURES

ASME TEST FORM PTC 32.1 PWR

For Capacity, Reactor Power Level, and Efficiency Tests
Nuclear Steam Supply System Using Pressurized Water Reactor

Plant ____________ Test No. ____________ Date ____________
Test Conducted By ____________ Location ____________ Unit No. ____________
Unit Make & Type ____________ Objective of Test ____________

Summary of Results:
- Duration ____________
- Capacity ____________
- Reactor Power Level ____________
- Efficiency ____________

	PARAMETER	LOCATION IN FIG. 1		
1.	Steam pressure at steam generator outlet	1	____	psia
2.	Steam temperature at steam generator outlet	1	____	F
3.	Steam flow from steam generator	1	____	lb/hr
4.	Steam quality at steam generator outlet	1	____	%
5.	Sat. liquid enthalpy at steam generator outlet	1	____	Btu/lb
6.	Sat. steam enthalpy at steam generator outlet	1	____	Btu/lb
7.	Steam enthalpy at steam generator outlet	1	____	Btu/lb
8.	Feedwater temperature	2	____	F
9.	Feedwater flow	2	____	lb/hr
10.	Enthalpy, feedwater	2	____	Btu/lb
11.	Blowdown temperature	3	____	F
12.	Blowdown flow	3	____	lb/hr
13.	Blowdown enthalpy	3	____	Btu/lb
14.	Reactor coolant letdown temperature	4	____	F
15.	Reactor coolant letdown flow	4	____	lb/hr
16.	Reactor coolant letdown enthalpy	4	____	Btu/lb
17.	Reactor coolant makeup temperature	5	____	F
18.	Reactor coolant makeup flow	5	____	lb/hr
19.	Reactor coolant makeup enthalpy	5	____	Btu/lb
20.	Seal water inlet temperature	10	____	F
21.	Seal water inlet flow	10	____	lb/hr
22.	Seal water inlet enthalpy	10	____	Btu/lb
23.	Seal water outlet temperature	9	____	F
24.	Seal water outlet flow	9	____	lb/hr
25.	Seal water outlet enthalpy	9	____	Btu/lb
26.	Cooling water inlet temperature	12	____	F
27.	Cooling water inlet flow	12	____	lb/hr
28.	Cooling water inlet enthalpy	12	____	Btu/lb
29.	Cooling water outlet temperature	13	____	F
30.	Cooling water outlet flow	13	____	lb/hr
31.	Cooling water outlet enthalpy	13	____	Btu/lb
32.	Pressurizer heater power	11	____	kw
33.	Reactor coolant pump power	6	____	kw
34.	Miscellaneous electrical inputs to envelope	7	____	kw
35.	Radiation and convection losses from envelope	8	____	Btu/hr
36.	Average air temperature inside containment		____	F
37.	Ambient temperature outside containment		____	F

N U C L E A R   S T E A M   S U P P L Y   S Y S T E M S

ASME TEST FORM PTC 32.1 BWR

For Capacity, Reactor Power Level, and Efficiency Tests
Nuclear Steam Supply System Using Boiling Water Reactor

Plant ______________________ Test No. ______________ Date __________
Test Conducted By ______________ Location ______________ Unit No. __________
Unit Make & Type ______________ Objective of Test ______________________

Summary of Results:
- Duration ______________________
- Capacity ______________________
- Reactor Power Level ______________________
- Efficiency ______________________

	PARAMETER	LOCATION IN FIG. 3		
1.	Steam pressure at reactor outlet	1	________	psia
2.	Steam temperature at reactor outlet	1	________	F
3.	Steam flow from reactor	1	________	lb/hr
4.	Sat. liquid enthalpy at reactor outlet	1	________	Btu/hr
5.	Sat. steam enthalpy at reactor outlet	1	________	Btu/hr
6.	Steam enthalpy at reactor outlet	1	________	Btu/hr
7.	Steam quality at reactor outlet	1	________	%
8.	Feedwater temperature	2	________	F
9.	Feedwater flow	2	________	lb/hr
10.	Feedwater enthalpy	2	________	Btu/lb
11.	Seal water inlet temperature	3	________	F
12.	Seal water inlet flow	3	________	lb/hr
13.	Seal water inlet enthalpy	3	________	Btu/lb
14.	Seal water outlet temperature	4	________	F
15.	Seal water outlet flow	4	________	lb/hr
16.	Seal water outlet enthalpy	4	________	Btu/lb
17.	Reactor cleanup system cooling water inlet temperature	7	________	F
18.	Reactor cleanup system cooling water flow	7	________	lb/hr
19.	Reactor cleanup system cooling water inlet enthalpy	7	________	Btu/lb
20.	Reactor cleanup system cooling water outlet temperature	8	________	F
21.	Reactor cleanup system cooling water outlet enthalpy	8	________	Btu/lb
22.	Blowdown outlet temperature	9	________	F
23.	Blowdown outlet flow	9	________	lb/hr
24.	Blowdown outlet enthalpy	9	________	Btu/lb
25.	Cooling water inlet temperature	5	________	F
26.	Cooling water inlet flow	5	________	lb/hr
27.	Cooling water inlet enthalpy	5	________	Btu/lb
28.	Cooling water outlet temperature	6	________	F
29.	Cooling water outlet flow	6	________	lb/hr
30.	Cooling water outlet enthalpy	6	________	Btu/lb
31.	Control rod drive water temperature	11	________	F
32.	Control rod drive water flow	11	________	lb/hr
33.	Control rod drive water enthalpy	11	________	Btu/lb
34.	Recirculating pump power input	12	________	kw
35.	Miscellaneous electrical inputs	13	________	kw
36.	Radiation and convection losses from envelope	10	________	Btu/hr
37.	Average temperature inside containment		________	F
38.	Ambient temperature outside containment		________	F

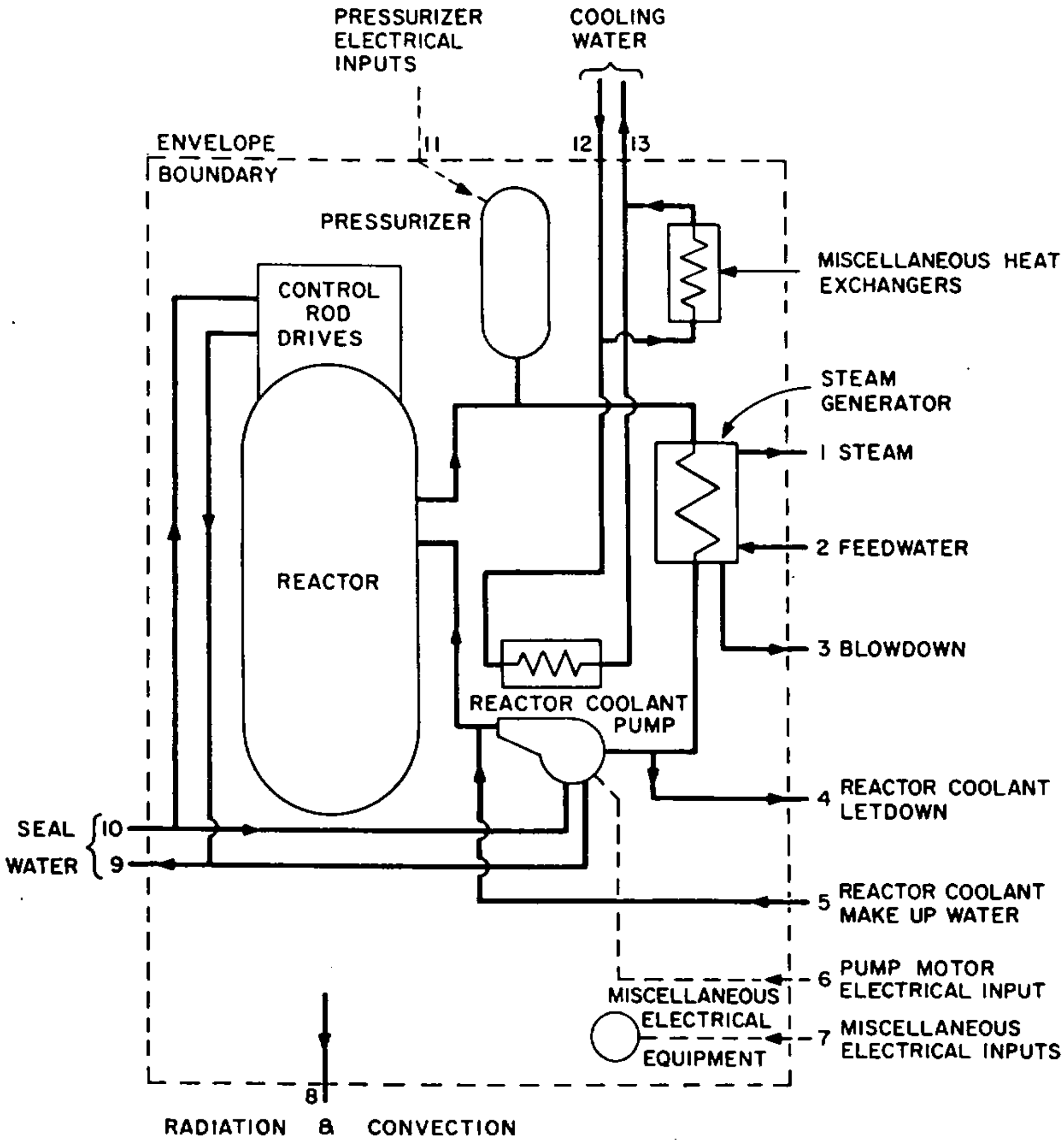

FIG. 1 NUCLEAR STEAM SUPPLY SYSTEM DIAGRAM – PRESSURIZED WATER REACTOR

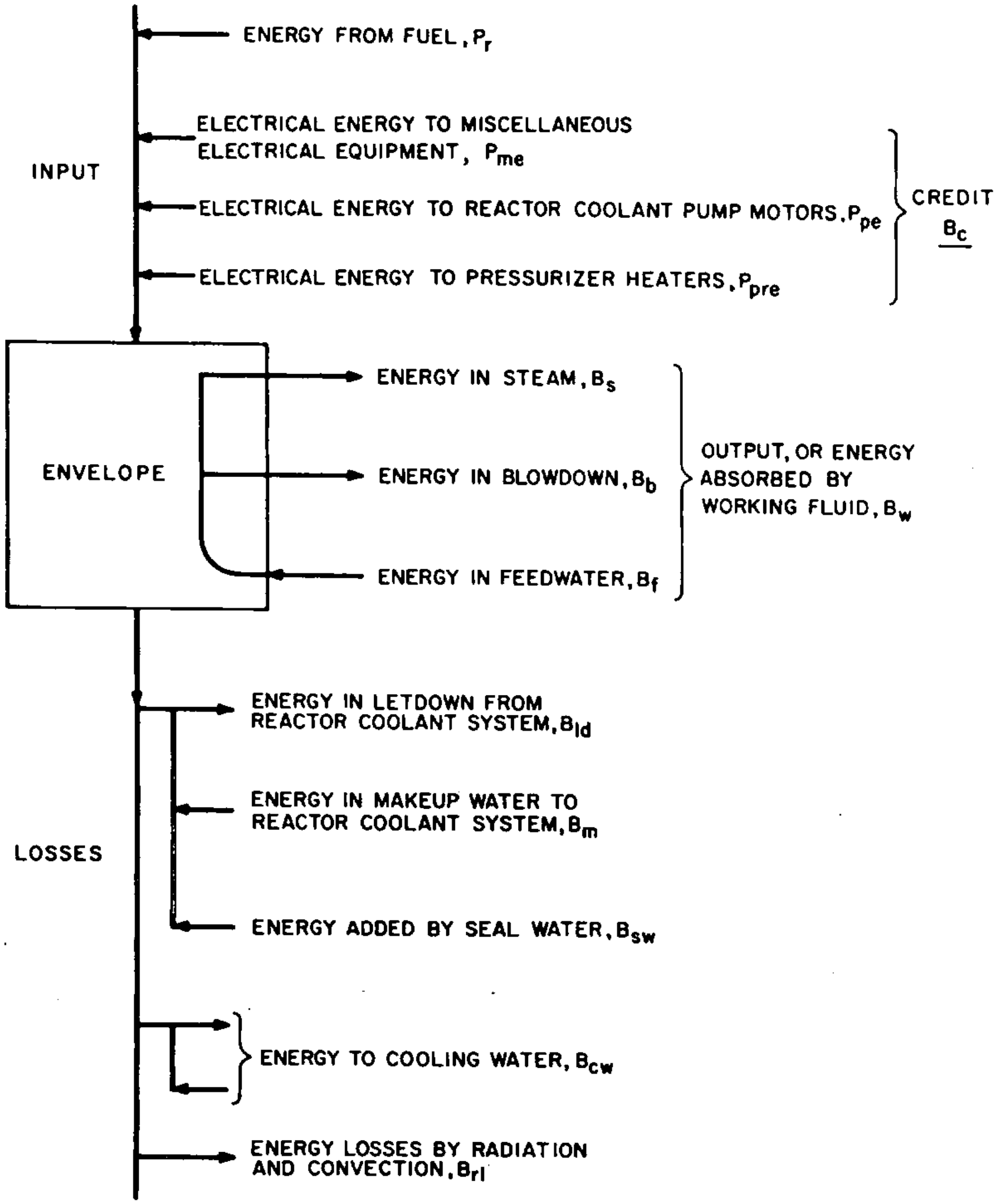

FIG. 2 ENERGY BALANCE OF NUCLEAR STEAM SUPPLY SYSTEM – PRESSURIZED WATER REACTOR

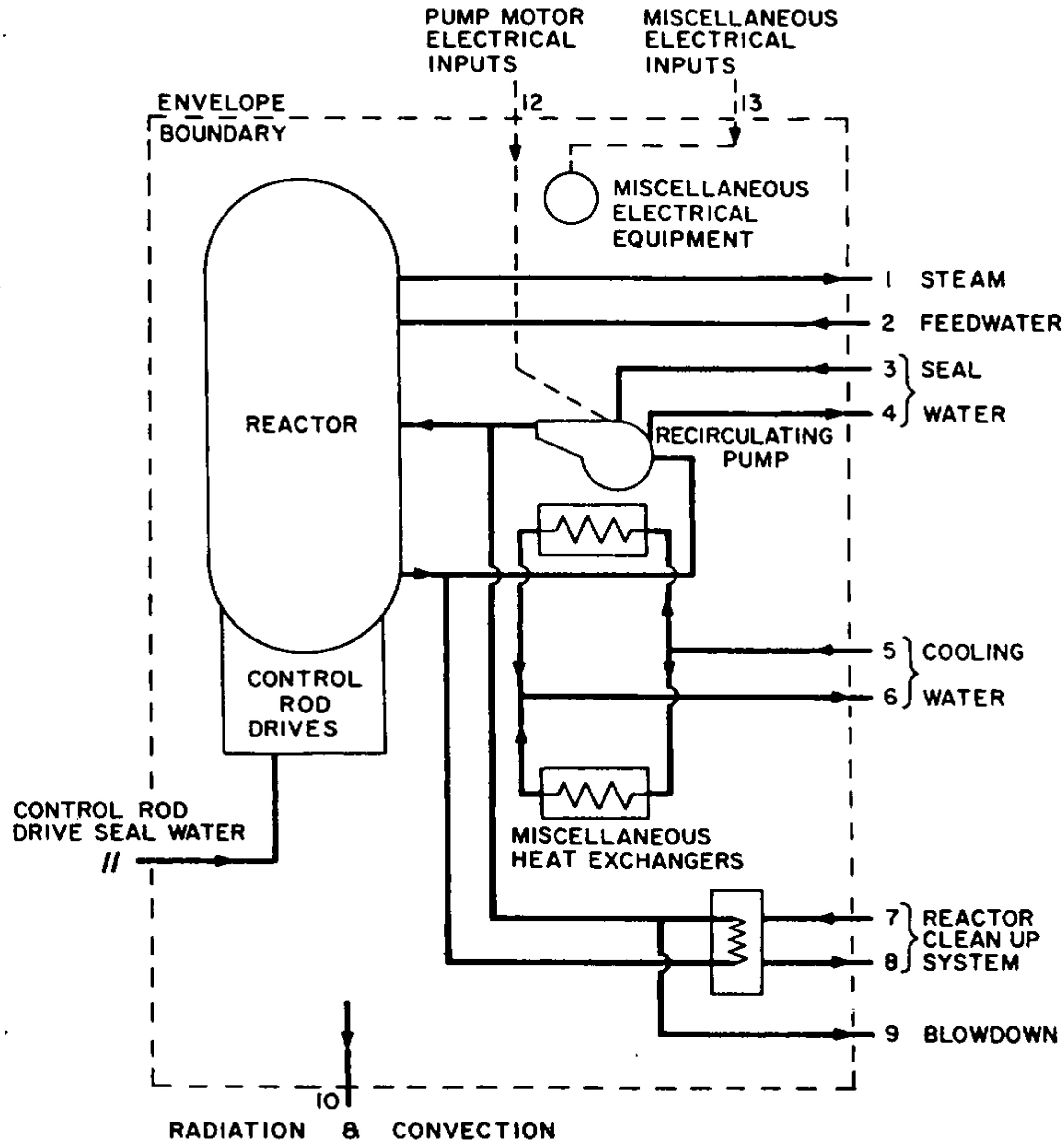

FIG. 3 NUCLEAR STEAM SUPPLY SYSTEM DIAGRAM – BOILING WATER REACTOR

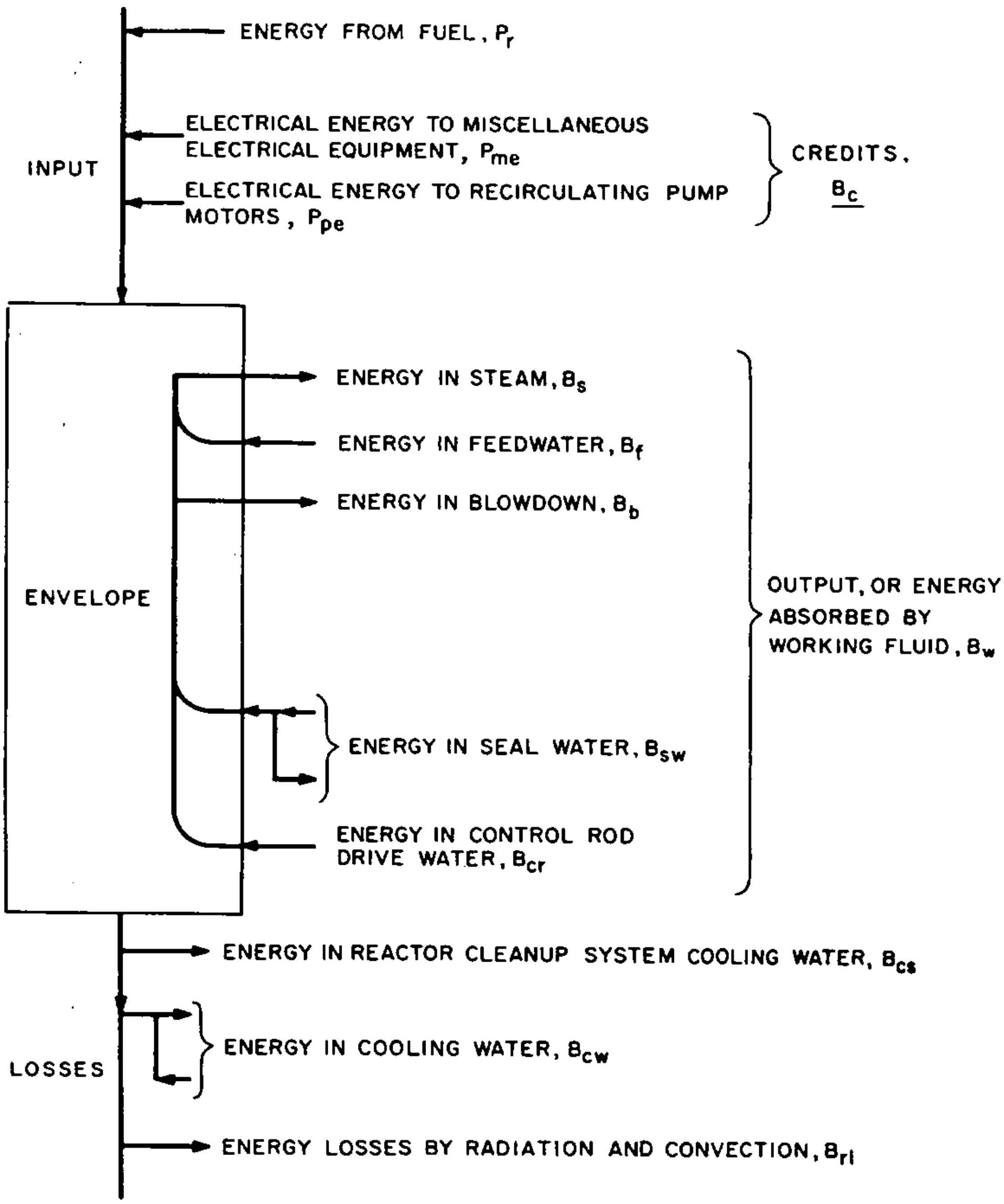

FIG. 4 ENERGY BALANCE OF NUCLEAR STEAM SUPPLY SYSTEM – BOILING WATER REACTOR

PERFORMANCE TEST CODES

While providing for exhaustive tests, these Codes are so drawn that selected parts may be used for tests of limited scope.

PERFORMANCE TEST CODES NOW AVAILABLE

PTC 23	– Atmospheric Water Cooling Equipment	(1958)
PTC 8.2	– Centrifugal Pumps	(1965)
PTC 4.2	– Coal Pulverizers	(1969)
PTC 1	– Code on General Instructions	(1945)
PTC 2	– Code on Definitions and Values	(1945)
PTC 10	– Compressor and Exhausters	(1965)
PTC 9	– Displacement Compressors, Vacuum Pumps and Blowers	(1954)
PTC 2.1	– Displacement Pumps	(1962)
PTC 12.3	– Deaerators	(1958)
PTC 27	– Determining Dust Concentration in a Gas Stream	(1957)
PTC 28	– Determining the Properties of Fine Particulate Matter	(1965)
PTC 3.1	– Diesel and Burner Fuels	(1958)
PTC 21	– Dust Separating Apparatus	(1941)
PTC 24	– Ejectors and Boosters	(1956)
PTC 14	– Evaporating Apparatus	(1955)
PTC 12.1	– Feedwater Heaters	(1955)
PTC 16	– Gas Producers and Continuous Gas Generators	(1958)
PTC 22	– Gas Turbine Power Plants	(1966)
PTC 3.3	– Gaseous Fuels	(1969)
PTC 18	– Hydraulic Prime Movers	(1949)
PTC 17	– Internal Combustion Engines	(1957)
PTC 32.1	– Nuclear Steam Supply Systems	(1969)
PTC 20.2	– Overspeed Trip Systems for Steam Turbine–Generator Units	(1965)
PTC 7	– Reciprocating Steam-Driven Displacement Pumps	(1949)
PTC 5	– Reciprocating Steam Engines	(1949)
PTC 25.2	– Safety and Relief Valves	(1966)
PTC 3.2	– Solid Fuels	(1954)
PTC 29	– Speed-Governing Systems for Hydraulic Turbine-Generator Units	(1965)
PTC 26	– Speed-Governing Systems for Internal Combustion Engine-Generator Units	(1962)
PTC 20.1	– Speed-Governing Systems for Steam Turbine-Generator Units	(1958)
PTC 12.2	– Steam Condensing Apparatus	(1955)
PTC 4.1	– Steam-Generating Units	(1964)
PTC 6	– Steam Turbines	(1964)
PTC 6A	– Appendix A to Test Code for Steam Turbines	(1964)
PTC 6	– Report on Guidance for Evaluation of Measurement Uncertainty in Performance Tests of Steam Turbines	(1969)