4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is, subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 252 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: “Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan,” submitted by letter dated May 19, 2006.

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1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs)

1.1.1 Reactor Core SLs

Applicability

This specification applies to the limiting combinations of reactor power and reactor coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

(a) The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures as shown in Figure 1-1 for 4-pump operation. The safety limit is exceeded if the point defined by the combination of reactor coolant cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

(b) Peak fuel centerline temperature shall be maintained at < 5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56(P)(A), Revision 1, Supplement 1.

1.1.2 Reactor Coolant System Pressure SL

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity to the containment.

Specification

The reactor coolant system pressure shall not exceed 2750 psia when fuel assemblies are located within the reactor vessel.

1.0 SAFETY LIMITS

1.1 Safety Limits (SLs) (continued)

Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At DNB there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through a correlation. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure 1-1 represent the loci of points for reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the minimum DNBR is not less than the minimum DNBR limit. The area of safe operation is below these lines.

SL 1.1.1(b) ensures that fuel centerline temperature remains below the fuel melt temperature 5081°F during normal operating conditions or design anticipated operational occurrences (AOOs) with adjustments for burnup and burnable poison. An adjustment of 58°F per 10,000 MWD/MTU has been established in XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5 (Ref. 8) and adjustments for burnable poisons are established based on XN-NF-79-56(P)(A), Revision 1, Supplement 1 (Ref. 9).

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2.10 and axial shapes within the axial power distribution trip limits in the COLR. The Thermal Margin/Low Pressure trip requirements shall be within the limits provided in the COLR. The Thermal Margin/Low Pressure trip is based on an unrodded integrated total radial peak (F_R^T) that is provided in the COLR.

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs) (continued)

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model test.[2] The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the minimum DNBR limit.[1]

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere.[3] In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of the reactor coolant system and fuel cladding. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[4]

The settings and capacity of the main steam safety valves (1000 - 1050 psia)[5], the reactor high-pressure trip (≤ 2400 psia) and the reactor coolant system safety valves (2500-2545 psia)[6] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test pressure was conducted at 3125 psia (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves, consistent with the reactor high pressure trip, and opening the steam system steam dump and bypass valves upon receipt of a turbine trip signal.[7]

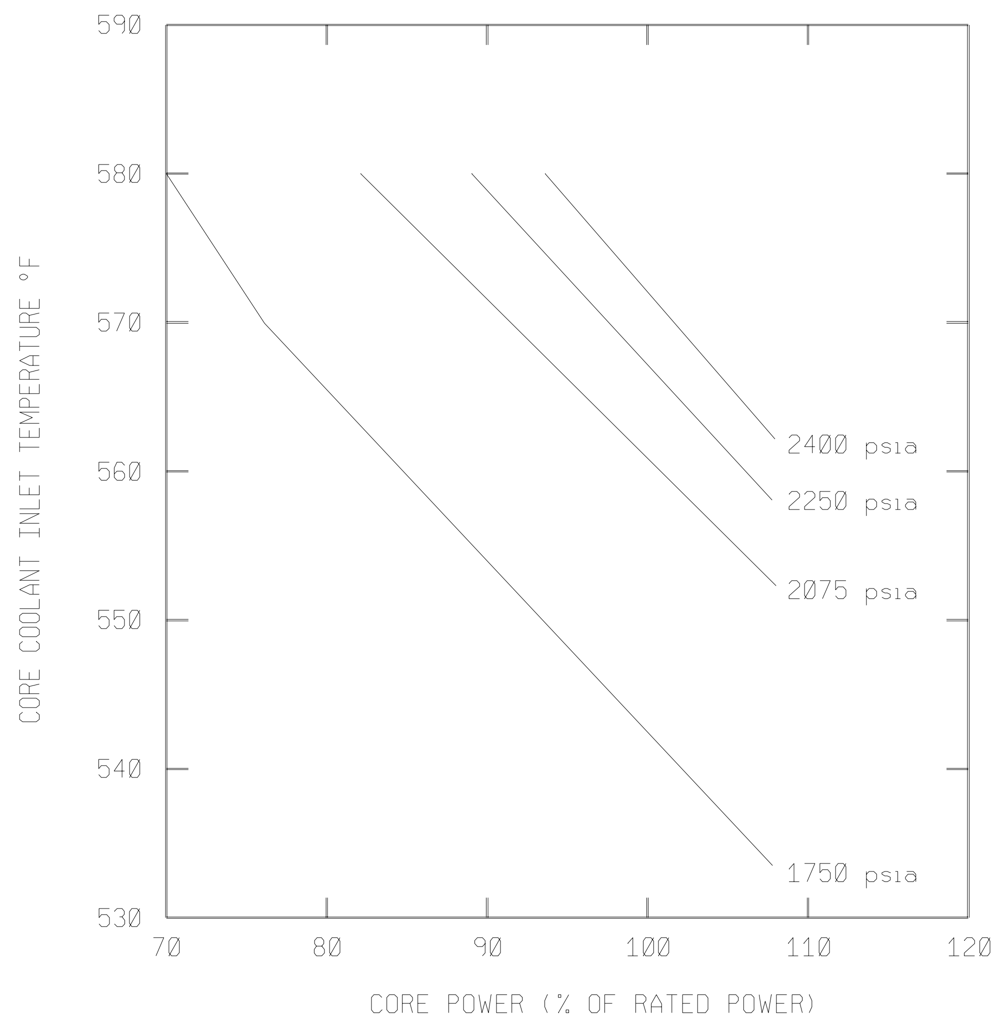
References

(1) USAR, Section 3.6.6
(2) USAR, Section 1.4.6
(3) USAR, Section 4
(4) USAR, Section 4.3.3
(5) USAR, Section 4.3.4
(6) USAR, Section 4.3.9.5
(7) USAR, Section 7.4.5.1
(8) XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” October 1986
(9) XN-NF-79-56(P)(A), Revision 1, Supplement 1, “Gadolinia Fuel Properties of LWR Fuel Safety Evaluation,” November 1981

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs) (continued)

Figure 1-1



Thermal Margin/Low Pressure Safety Limits 4 Pump Operation	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 1-1

1.0 **SAFETY LIMITS**

1.2 Safety Limit Violations

1.2.1 If Safety Limit 1.1.1 is violated, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2 If Safety Limit 1.1.2 is violated:

1.2.2.1 In MODE 1 or 2, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2.2 In MODES 3, 4, or 5, restore compliance within 5 minutes.

Basis

SL 1.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be operable during MODES 1 and 2 to ensure operation within the reactor core SLs. Applicability is not required in those operating MODES when the reactor is not generating significant thermal power.

If SL 1.1.1 is violated, the requirement to go to HOT SHUTDOWN places the unit in a MODE in which the SL is not applicable. The allowed completion time of 1 hour recognizes the importance of bringing the unit to a MODE where this SL is not applicable and reduces the probability of fuel damage.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in HOT SHUTDOWN within 1 hour. The allowed completion time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions and to establish HOT SHUTDOWN conditions.

If the RCS pressure SL is exceeded in MODES 3, 4, or 5, RCS pressure must be restored within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODES 3, 4 or 5 is potentially more severe than exceeding this SL in MODES 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System

Applicability

This specification applies to RPS Limiting Safety System settings and bypasses for instrument channels.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be within the Limiting Safety System Setting as stated in Table 2-11.

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

(1) High Power Level - A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from some reactivity excursions too rapid to be detected by pressure and temperature measurements (in addition, thermal signals are provided to the high power level trip unit as a backup to the neutron flux signal).

During normal plant operation, with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 109.0% of indicated full power. Adding to this the possible variation in trip point due to calibration and measurement errors, the maximum actual steady-state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.(1)

During reactor operation at power levels between 19.1% and 100% of rated power, the Variable High Power Trip (VHPT) will initiate a reactor trip in the event of a reactivity excursion that increases reactor power by 10% or less of rated power. The high power trip setpoint can be set no more than 10% of rated power above the indicated plant power. Operator action is required to increase the set point as plant power is increased. The setpoint is automatically decreased as power decreases.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

During reactor operation at power levels below 19.1% rated power, a reactor trip will occur in the event of a reactivity excursion that results in a power increase up to the lower fixed set point of the VHPT circuit of 19.1% of rated power.(3) During normal power increases below 19.1% reactor trip would be initiated at 19.1% of rated power unless the set point is manually adjusted.

(2) Low Reactor Coolant Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.

Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves(2).

During four-pump operation, the low flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% (5) of the assumed minimum indicated value. Refer to Technical Specification 2.10.4 (5)(a)(iii) for the minimum indicated value of reactor coolant flow.

(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.(1)

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than the minimum DNBR limit, including allowance for measurement error. The thermal and hydraulic limits shown in the Thermal Margin/Low Pressure 4-Pump Operation Figure, contained in the COLR, define the limiting values of reactor coolant pressure, reactor inlet temperature, axial shape index, and reactor power level which ensure that the thermal criteria(8) are not exceeded. The low set point of 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the equation given in the COLR for the Thermal Margin/Low Pressure Limit.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

(5) Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feedwater flow accidents and assures that the design pressure of the reactor coolant system will not be exceeded. The specified set points assure that there will be sufficient water inventory in the steam generator at the time of trip to provide a 12-minute margin before the auxiliary feedwater is required.[9]

The setting listed in Table 2-11 assures that the heat transfer surface (tubes) are covered with water when the reactor is critical.

(6) Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 770 psia so as not to interfere with normal operation but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ±22 psia in the accident analysis.[1]

(7) Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment high pressure signal which indicates safety injection system operation.

(8) Axial Power Distribution - The axial power trip is provided to ensure that excessive axial peaking will not cause fuel damage. The Axial Shape Index is determined from the axially split excore detectors. The set point functions, shown in the COLR ensure that neither a DNBR of less than the minimum DNBR limit nor a fuel centerline temperature greater than the safety limit corresponding to Fuel Centerline Melt (FCM), as determined each fuel cycle and contained in the COLR, will exist as a consequence of axial power maldistributions. Allowances have been made for instrumentation inaccuracies and uncertainties associated with the excore symmetric offset - incore axial peaking relationship. A variance of 5% between Δ T-Power and NI-Power is permitted due to the significant margins to local power density limits before calibration of NI-Power is performed at 30% power.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than the minimum DNBR limit nor a fuel centerline temperature greater than the safety limit corresponding to FCM, as determined each fuel cycle and contained in the COLR, occurs as a result of the loss of load to one steam generator.

(10) Physics Testing at Low Power - During physics testing at power levels less than 10^{-1}% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below 10^{-1}% of rated power. Written test procedures which are approved by the Plant Review Committee will be in effect during these tests. At reactor power levels less than 10^{-1}% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6.
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6
(9) USAR, Section 14.10

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.13 Limiting Safety System Settings, Reactor Protective System (continued)

TABLE 2-11

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Trip Setpoints
1	High Power Level (A) 4-Pump Operation	≤ 109.0% of Rated Power
2	Low Reactor Coolant Flow (B)(F) 4-Pump Operation	≥ 95% of 4-Pump Flow
3	Low Steam Generator Water Level	31.2% of Scale
4	Low Steam Generator Pressure (C)	≥ 500 psia
5	High Pressurizer Pressure	≤ 2400 psia
6	Thermal Margin/Low Pressure (B)(F)	1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in the Thermal Margin/Low Pressure 4-Pump Operation Figure provided in the COLR)
7	High Containment Pressure (D)	≤ 5 psig
8	Axial Power Distribution (E)	(as shown in the Axial Power Distribution for 4-Pump Operation Figure provided in the COLR)
9	Steam Generator Differential Pressure	≤ 135 psid

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B May be bypassed below 10^{-4}% power.

C May be bypassed below 600 psia.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than 10^{-1}% of rated power the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of 10^{-1}% of rated power is operable.

5.0 **ADMINISTRATIVE CONTROLS**

5.7 Not used.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.21.

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

5.0 **ADMINISTRATIVE CONTROLS**

5.9 Reporting Requirements (Continued)

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Not Used

5.9.5 Core Operating Limits Report (COLR)

a. Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.2.7	Borated Water Sources - Shutdown
2.2.8	Borated Water Sources - Operating
2.10.2	Reactivity Control Systems and Core Physics Parameters Limits
2.10.4	Power Distribution Limits
2.13	RPS Limiting Safety System Settings, Table 2-11, Items 6, 8, and 9

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: