AREVAEPRDCPEm Resource

From: BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent: Monday, November 01, 2010 4:15 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); RYAN Tom (AREVA); HOLM Jerald (EXTERNAL AREVA); HALLINGER Pat (EXTERNAL AREVA)
Subject: DRAFT Response to U.S. EPR Design Certification Application RAI No. 432, FSAR Ch. 15, OPEN ITEM, Supplement 1
Attachments: RAI 432 Response -DRAFT.pdf

Getachew,

Attached is a draft response to support the final response date of November 30, 2010. Let me know if the staff has questions or if this can be sent as final.

Thanks

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (External RS/NB)
Sent: Wednesday, October 27, 2010 11:19 AM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); HOLM Jerald (External RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 432, FSAR Ch. 15, OPEN ITEM, Supplement 1

Getachew,

AREVA NP provided a schedule for a technically correct and complete response to RAI 432 on September 15, 2010. The schedule is being revised as shown below.

AREVA NP's schedule for providing a technically correct and complete response to the question in RAI 432 is provided below.

Question #	Response Date
RAI 432 — 15.00.02-1	November 30, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: BRYAN Martin (External RS/NB)
Sent: Wednesday, September 15, 2010 11:55 AM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); HOLM Jerald (External RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 432, FSAR Ch. 15, OPEN ITEM

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 432 Response US EPR DC.pdf" provides a schedule for a response to the single question in this RAI.

The schedule for a technically correct and complete response to this question is provided below.

Question #	Response Date
RAI 432 – 15.00.02-1	October 29, 2010

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, August 16, 2010 12:33 PM
To: ZZ-DL-A-USEPR-DL
Cc: Liang, Chu-Yu; Lu, Shanlai; Donoghue, Joseph; Carneal, Jason; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 432 (4958), FSAR Ch. 15, OPEN ITEM

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on August 6, 2010, and on August 16, 2010, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 2219

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB71081F1E92)

Subject: DRAFT Response to U.S. EPR Design Certification Application RAI No. 432, FSAR Ch. 15, OPEN ITEM, Supplement 1
Sent Date: 11/1/2010 4:15:05 PM
Received Date: 11/1/2010 4:17:37 PM
From: BRYAN Martin (EXTERNAL AREVA)

Created By: Martin.Bryan.ext@areva.com

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Tracking Status: None
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Tracking Status: None
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Tracking Status: None

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Files	**Size**	**Date & Time**
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RAI 432 Response -DRAFT.pdf		472466

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 432(4958), Revision 1

8/16/2010

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.00.02 - Review of Transient and Accident Analysis Methods
01/2006
Application Section: 15.0.0.3.1

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

Question 15.00.02-1:

OPEN ITEM:

In EPR FSAR Tier 2, Revision 1, Section 15.0.0.3.1, ”Design Plant Conditions and Initial Conditions”, indicated that a heat balance measurement uncertainty of +/- 22 MWt is applicable to the rated core thermal power of 4590 MWt (approximately 0.48% uncertainty) for the maximum core power assumed in the accident analyses. The core power is determined using a secondary-side heat balance. The relatively low heat balance uncertainty is achieved by using an ultrasonic flow meter for the feedwater flow rate. The maximum power level assumed in the accident analyses are described in Table 15.0.4, ”Nuclear Steam Supply System Power Levels Assumed in the Accident Analyses” and Table 15.0.5, ”Plant Parameters Used in Accident Analyses” lists the nominal plant parameters for the accident analyses. Uncertainties in initial plant conditions are applied in accordance with the applicable approved methodologies.

However, EPR FSAR Revision 1 does not provide sufficient information of the instrumentation and/or methodology for the main feedwater flow measurement, nor provide a basis for the statement that the main feedwater flow measurement supports a 0.48% power uncertainty.

Please provide the following:

1) Describe the mechanism, such as the EPR FSAR and ITAAC and/or a COL Action Item, by which the information will be provided to support claimed 0.48% power measurement uncertainty and how it will be verified and confirmed.

2) The following information should be provided to support the claimed 0.48% power measurement uncertainty:

   A. A description of the instrumentation and methodology used for the main feedwater flow measurement and calorimetric power measurement.

   B. All of the following:

      a. A reference to the NRC approval of the main feedwater and power measurement methodology, instrumentation, and associated uncertainties. Or

      b. The instrument string, including applicable sensors or transducers, process rack, analog/digital converter, process computer, and readout devices, etc., for each parameter measured;

      c. The accuracy of allowance associated with each instrument component, such as sensor reference, calibration, and measurement accuracies, respectively; rack calibration and measurement accuracies; sensor pressure and temperature effects; rack pressure and temperature effects; drift; process measurement accuracy; instrument range, span, and operation limits, etc.;

      d. The methodology for combining uncertainties, allowances, or errors of the instrument components associated with each parameter to arrive at the overall uncertainty of each measured parameter; and

e. The methodology used to arrive at the total uncertainties for the main feedwater flow rate and reactor thermal power, respectively.

Response to Question 15.00.02-1:

1) The 0.48 percent power measurement uncertainty is verified by a calculation performed in accordance with Reference 1 and Reference 2. Reactor power is confirmed by a continuous secondary side calorimetric which uses input that meets the uncertainty requirements specified in Table 15.00.02-1-1. The U.S. EPR FSAR Tier 2, Section 7.7.2.3.5 will be revised to provide additional details regarding the secondary side calorimetric power measurement.

2) The U.S. EPR reactor control, surveillance, and limitation system (RCSL) implements a continuous secondary side calorimetric that monitors and limits core power. This limitation function, or the reactor power limitation with respect to thermal power, is addressed in U.S. EPR FSAR Tier 2, Section 7.7.2.3.5.

A. During steady state operation, this function relies on the maximum value of the continuous secondary side calorimetric and the RCS enthalpy indication of reactor power as input. The RCS enthalpy indication of reactor power is calibrated to the secondary side calorimetric during 100 percent steady state conditions. Although the RCS enthalpy indication of reactor power has more uncertainty than the secondary side calorimetric, its uncertainty is smaller than that of the power range ex-core detectors, with less signal noise. Unlike the power range ex-core detectors, the RCS enthalpy indication of reactor power is not decalibrated by changes in downcomer temperature. U.S. EPR FSAR Tier 2, Section 7.7.2.3.5 will be revised by removing the power range ex-core detector indication of reactor power as an input to this limitation function.

The continuous secondary side calorimetric uses the following sensors and parameters as input:

– Feedwater flow rate for each train of feedwater (refer to U.S. EPR FSAR Tier 2, Section 10.4.7.5 and Figure 10.4.7-1).

– Feedwater temperature for each train of feedwater (refer to U.S. EPR FSAR Tier 2, Section 10.4.7.5 and Figure 10.4.7-1).

– Feedwater pressure for each train of feedwater (U.S. EPR FSAR Tier 2, Figure 10.4.7-1 will be revised to show the pressure sensors).

– Steam generator blowdown flow rate for each steam generator (refer to U.S. EPR FSAR Tier 2, Figure 10.4.8-1).

– Steam generator blowdown temperature for each steam generator (refer to U.S. EPR FSAR Tier 2, Figure 10.4.8-1).

– Reactor coolant system charging flow rate (refer to U.S. EPR FSAR Tier 2, Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).

- Reactor coolant system charging flow temperature (refer to U.S. EPR FSAR Tier 2, Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).

- Reactor coolant system charging flow pressure (refer to U.S. EPR FSAR Tier 2, Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).

- Reactor coolant system letdown flow rate from both high pressure reducing stations in the chemical and volume control system (refer to U.S. EPR FSAR Tier 2, Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).

- Reactor coolant system letdown flow temperature (refer to U.S. EPR FSAR Tier 2, Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).

- Reactor coolant system letdown flow pressure (a constant value is assumed).

- Main steam pressure for each steam generator (refer to U.S. EPR FSAR Tier 2, Figure 10.3-1).

- The power losses from the reactor coolant system (including the control rod drive mechanisms) to the ambient air (a constant value is assumed).

- The reactor coolant pump power (a constant value is assumed).

- The pressurizer heater power (a constant value is assumed).

- The moisture content of the main steam (a constant value is assumed).

Note that the enthalpies of the main steam flow, main feedwater flow, steam generator blowdown flow, charging flow, and letdown flow are calculated using the corresponding pressures and/or temperatures.

The algorithm for the continuous secondary calorimetric calculation of reactor thermal power is performed according to methodology outlined in Reference 1 and approved by NRC in Reference 2.

B.

a. The methodology outlined in the response to Question 2A has been approved by the NRC (see Reference 2). As an analytical requirement, 0.48 percent uncertainty on core thermal power was assumed in the safety analysis. However, the measurement requirements for the U.S. EPR allow the secondary side calorimetric to calculate reactor thermal power within a ± 0.40 percent uncertainty. In order to achieve the required uncertainty in the secondary side calorimetric algorithm, the elemental uncertainties of the instrument strings and parameters previously mentioned are provided in Table 15.00.02-1-1.

Table 15.00.02-1-1—Elemental Uncertainties of the Instrument Strings and Parameters

Input	Maximum Allowable Uncertainty at 100% NP
Feedwater Flow Rate	0.28% of the Actual Value
Feedwater Temperature	± 0.6°F of the Actual Value
Feedwater Pressure	± 25 psia of the Actual Value
Steam Pressure	± 25.4 psia of the Actual Value
Blowdown Flow Rate	± 5% of the Actual Value
Blowdown Temperature	± 3.0°F of the Actual Value
Charging Flow Rate	± 4% of the Actual Value
Charging Temperature	± 3% of the Actual Value
Charging Pressure	± 3% of the Actual Value
Letdown Flow Rate	± 4% of the Actual Value
Letdown Temperature	± 3% of the Actual Value
Letdown Pressure	± 3% of the Actual Value
Reactor Coolant Pump Power	± 20% of the Actual Value
Power Losses from the Reactor Coolant System	± 20% of the Actual Value
Pressurizer Heater Power	± 20% of the Actual Value
Steam Moisture Content	± 0.25% of the Actual Value

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 7.7.2.3.5 and Figure 10.4.7-1 will be revised as described in the response and indicated on the enclosed markup.

References:

1. Engineering Report ER-157P, Topical Report, Revision 8, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System”, Cameron Measurement Systems.

2. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Engineering Report ER-157P, Topical Report, Revision 8, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,” Cameron Measurement Systems,” Project No. 1370.

U.S. EPR Final Safety Analysis Report Markups

Loss of All MFW Pumps

A low MFW flowrate combined with a high reactor power level is the criteria for the detection of the loss of all MFW pumps. In this case the limitation function will initiate a non-safety-related reactor trip, activate turbine trip, and close all FW FLCVs. The reactor trip signal resets this actuation.

Imbalance of Feedwater Flowrate and Reactor Power During Startup Phase

Indications of a low enough feedwater flowrate and a high enough reactor power leads to blocking the withdrawal of any RCCA. This prevents an increase of the reactor power without an increase of the MFW flowrate during the startup phase.

7.7.2.3.4 Reactor Power Limitation with respect to Generator Power

This limitation function limits reactor power after loss of generator load events. The objective is to limit the energy level of the primary system in case of load rejections or turbine trip in order to avoid reaching the RT criteria. This will be done by initiating a PT. The target reactor power level is determined by:

- The maximum of generator power.
- The minimum PT target power.

In case of turbine trip or load rejection to house load, the plant is first stabilized at minimum PT target power while heat removal is performed via the turbine bypass valves. A further controlled reduction to the minimum load reactor power will then be done by ACT control.

7.7.2.3.5 Reactor Power Limitation with respect to Thermal Power

The reactor power limitation with respect to thermal power function is designed to maintain reactor power below 100 percent rated thermal power. This function provides the capability to adjust turbine power and indirectly reactor power due to cooling tower temperature changes that affect overall plant efficiencies. The reactor power signal is selected from the highest of the following:

15.00.02-1

- Continuous secondary calorimetric calculation (i.e., above 25 percent power).
- ~~Median select excore power range indication of reactor power.~~
- Median select RCS enthalpy indication of reactor power.

The continuous secondary side calorimetric uses the following sensors and parameters as input:

15.00.02-1

- Feedwater flow rate for each train of feedwater (refer to Section 10.4.7.5 and Figure 10.4.7-1).
- Feedwater temperature for each train of feedwater (refer to Section 10.4.7.5 and Figure 10.4.7-1).
- Feedwater pressure for each train of feedwater (refer to Figure 10.4.7-1).
- Steam generator blowdown flow rate for each steam generator (refer to Figure 10.4.8-1).
- Steam generator blowdown temperature for each steam generator (refer to Figure 10.4.8-1).
- Reactor coolant system charging flow rate (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system charging flow temperature (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system charging flow pressure (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system letdown flow rate from both high pressure reducing stations in the chemical and volume control system (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).
- Reactor coolant system letdown flow temperature (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).
- Reactor coolant system letdown flow pressure (a constant value is assumed).
- Main steam pressure for each steam generator (refer to Figure 10.3-1).
- The power losses from the reactor coolant system (including the control rod drive mechanisms) to the ambient air (a constant value is assumed).
- The reactor coolant pump power (a constant value is assumed).
- The pressurizer heater power (a constant value is assumed).
- The moisture content of the main steam (a constant value is assumed).

The enthalpy of the main steam flow, main feedwater flow, steam generator blowdown flow, charging flow, and letdown flow are calculated using the corresponding pressures and/or temperatures. The continuous secondary calorimetric calculation of reactor thermal power is performed according to methodology outlined in Reference 3, which has been accepted by the NRC, per Reference 4. As an analytical requirement, 0.48 percent uncertainty on core thermal power was assumed in the safety analysis. However, the measurement requirements for the U.S. EPR

15.00.02-1

allow the secondary side calorimetric to caluculate reactor thermal power within a ± 0.40 percent uncertainty. To achieve the required uncertainty in the secondary side calorimetric algorithm, the elemental uncertainties of the instrument strings and parameters, previously mentioned, are verified to comply with requirements provided in Table 7.7-1, "Elemental Uncertainties for Secondary Side Calorimetric."

The control logic compares the mismatch between main turbine and generator load and the highest of the previously listed power signals and takes actions when reactor power exceeds 100 percent. There are two thresholds. The intent of the first is to alert the operator and take action to prevent further power increase. The intent of the second threshold is to reduce power to 100 percent.

7.7.2.3.6 Rod Drop Limitation

The objective of this limitation function is to detect the spurious drop of RCCAs and to reduce the turbine generator power level to match the reactor power reduction due to the dropped RCCAs.

This limitation function is designed to avoid reactivity compensation by core control functions after the RCCAs drop and to avoid the low departure from nucleate boiling (DNBR) and high linear power density (HLPD) protective actuations after one or more RCCAs drop into the core.

Rod drop is detected in the protection system (PS) based on the RCCA position measurements. In each PS division a quarter of the RCCAs are monitored. Four (i.e., one per PS division) RCCA drop detection logic signals are acquired in RCSL and voted one out of four.

The other criterion indicating an RCCA drop is derived from the decrease of the reactor power level (i.e., neutron flux from power range detectors). The derivative of the four nuclear power signals are compared with a low threshold and voted one out of four.

The limitation will be actuated if both criteria coincide and no intended PT has been initiated by other limitation functions.

7.7.2.3.7 Intermediate Range High Neutron Flux Limitation

This limitation function is designed to avoid the high neutron flux (i.e., intermediate range) and low doubling time (i.e., intermediate range) reactor trips when an excessive reactivity increase occurs during reactor startup from a subcritical or a low power startup condition. At the limitation criteria the withdrawal of any RCCA is blocked.

Each RCSL division receives four binary limitation signals (i.e., one per PS division). Each limitation signal from the PS combines the following criteria:

7.7.2.12 Potential for Inadvertent Actuation

The non-safety control systems and functions are designed to limit the potential for inadvertent actuation and challenges to the safety systems. Many of the limitation I&C functions described in Section 7.7.2.3 are designed to achieve optimum plant availability. These types of limitation functions act before protection functions, and thus restore normal operating conditions without challenging the protection thresholds for the most frequent accident conditions. The limitation thresholds are set before protection thresholds (as close as possible to them), but with a margin taking into account the counter-measure response time. After exceeding a limitation threshold, rapid corrective actions are automatically initiated. The typical limitation function action is the RCCA dropping called the PT which leads to a fast power decrease.

7.7.2.13 Control of Access

Physical access to I&C cabinets is restricted to authorized personnel. Unauthorized electronic access to system software via network connections is prevented by administrative controls. The loading of software and parameter changes via maintenance equipment is only possible in accordance with clearly defined procedures.

7.7.3 Analysis

The control systems described in Section 7.7.1.1 and Section 7.7.2.1 are those used for normal operation that are not relied upon to perform safety functions following AOOs or accidents. These systems control plant processes having an impact on plant safety.

The plant control systems are designed to prevent an undesirable condition in the operation of the plant that, if reached, is protected by the PS. The description and analysis of this protection is covered in Section 7.2 and Section 7.3.

How these control systems comply with the acceptance criteria and conform to guidelines set forth in NUREG-0800 (Reference 2) is described in Section 7.1.

7.7.4 References

1. ANP-10304, Revision 1, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report," AREVA NP Inc., December 2009.

2. NUREG-0800, Standard Review Plan, Section 7.7, "Control Systems," Revision 5, U.S. Nuclear Regulatory Commission, March 2007.

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3. ER-157P, Topical Report, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," Cameron Measurement Systems.

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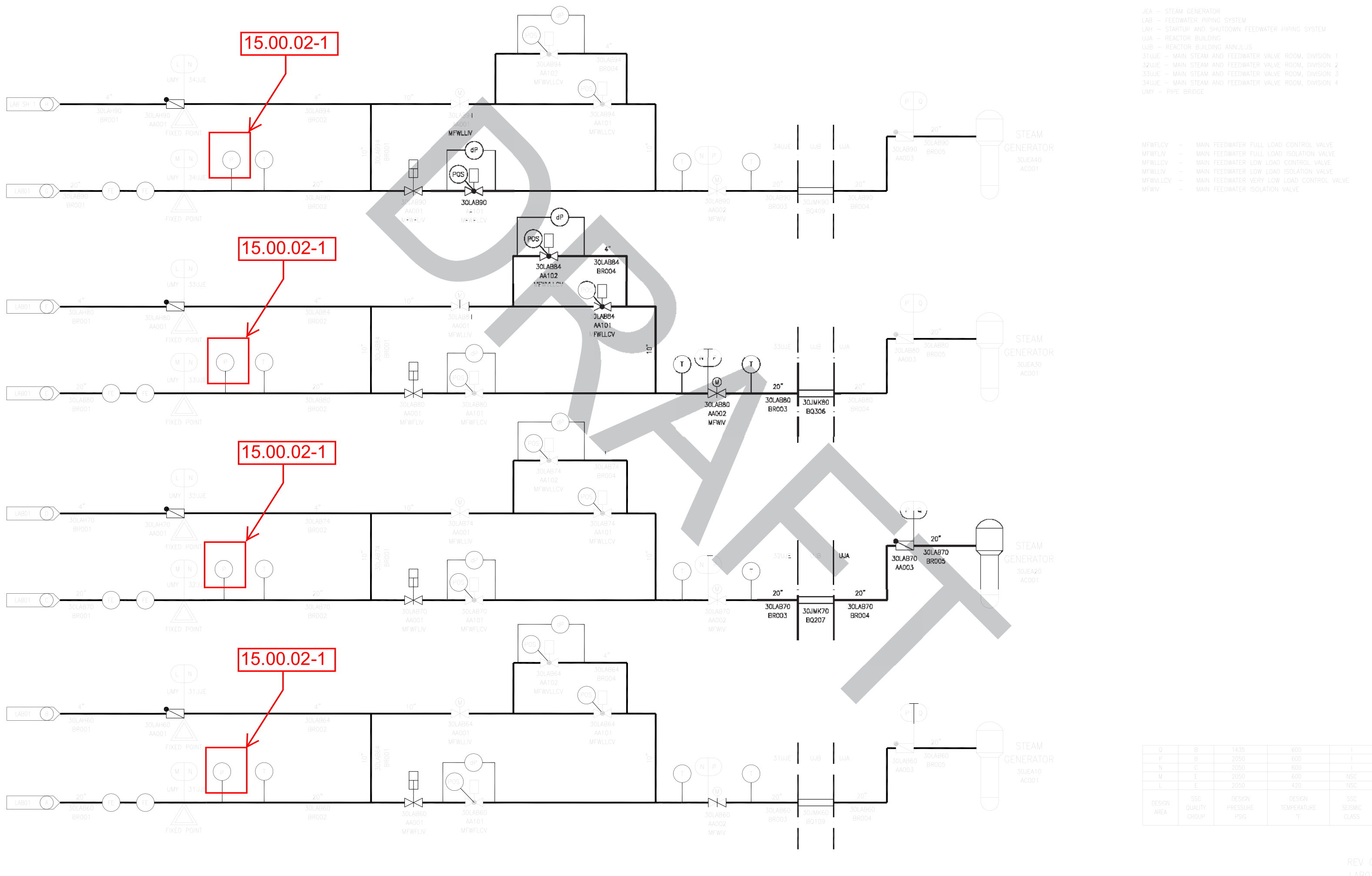
4. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Engineering Report ER-157P, Topical Report, Revision 8, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,” Cameron Measurement Systems,” Project No. 1370.

Table 7.7-2—Elemental Uncertainties for Secondary Side Calorimetric

Input	Maximum Allowable Uncertainty at 100% NP
Feedwater Flow Rate	0.28% of the Actual Value
Feedwater Temperature	± 0.6°F of the Actual Value
Feedwater Pressure	± 25 psia of the Actual Value
Steam Pressure	± 25.4 psia of the Actual Value
Blowdown Flow Rate	± 5% of the Actual Value
Blowdown Temperature	± 3.0°F of the Actual Value
Charging Flow Rate	± 4% of the Actual Value
Charging Temperature	± 3% of the Actual Value
Charging Pressure	± 3% of the Actual Value
Letdown Flow Rate	± 4% of the Actual Value
Letdown Temperature	± 3% of the Actual Value
Letdown Pressure	± 3% of the Actual Value
Reactor Coolant Pump Power	± 20% of the Actual Value
Power Losses from the Reactor Coolant System	± 20% of the Actual Value
Pressurizer Heater Power	± 20% of the Actual Value
Steam Moisture Content	± 0.25% of the Actual Value

15.00.02-1

Sheet 2 of 2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