February 9, 2006

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - REQUEST FOR ADDITIONAL
             INFORMATION (TAC NO. MC4570)

Dear Mr. Parrish:

By letter dated September 30, 2004, Energy Northwest submitted a request for license
amendment to Facility Operating License No. NPF-21 related to the application of Alternate
Source Term to the Columbia Generating Station. The Nuclear Regulatory Commission (NRC)
staff has performed a review of the amendment request and finds that it needs additional
information to complete its review.

Therefore, it is requested that you respond to the enclosed request for additional information
within 30 days of the date of this letter for the NRC staff to expedite its review. The enclosed
questions are unchanged, except for administrative changes, from those sent by e-mail to a
member of your staff on January 20, 2006.

                              Sincerely,

                              /RA/

                              Brian Benney, Project Manager
                              Plant Licensing Branch IV
                              Division of Operating Reactor Licensing
                              Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO ALTERNATIVE SOURCE TERM LICENSE AMENDMENT REQUEST

COLUMBIA GENERATING STATION

DOCKET NO. 50.397

Note: Unless otherwise specified, references to attachments and pages are to be considered from the September 30, 2004, submittal.

1.   Attachment 2, page 2, Table 1 states that the fission product inventory is ORIGEN 2-based. Please explain what ORIGEN 2-based means. Was the inventory determined using the ORIGEN 2 code?

2.   The main steam line break (MSLB) accident uses an assumption of 6 seconds for the maximum time for the main steam isolation valve (MSIV) closure. Please confirm that this assumption is to be used only for the purpose of radiological analysis for the MSLB calculation and not as a justification for changing technical specification (TS) surveillance closure times.

3.   Regulatory Guide (RG) 1.183, Position 5.1.2, "Credit for Engineered Safeguard Features," states:

> Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Attachment 2, Page 9 of the Columbia Generating Station (CGS) license amendment states that credited mitigation features meet these requirements and are automatic except residual heat removal (RHR) drywell sprays and standby liquid control injection. Please verify that RHR drywell sprays will be operable by TSs and are powered by emergency power sources.

4.   Attachment 1, page 51 of 91 states:

> A sensitivity calculation was performed to evaluate the significance of the dose contribution from [engineered safety

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features (ESFs)] leakage to the [condensate storage tank (CST)]. The calculation is included in the [alternative source term (AST) loss-of-coolant accident (LOCA)] analysis in Attachment 5. The dose contribution from the CST is negligible. The impact to the 30-day [low population zone] dose is less than 2 percent.

a) The LOCA analysis does not contain an Attachment 5. Was this reference to Attachment 2 of NE-02-04-05?

b) Please provide the RADTRAD input and outputs for the CST analysis.

c) Attachment 2 of NE-02-04-05 states that the liquid leakage to the CST is 0.48 gallons per minute (gpm). If the ESF leakage assumed is 1 gpm and is doubled to 2 gpm for the analysis, why is the CST leakage not 2 gpm?

d) Please justify not including this dose contribution since this pathway contributes to the total dose (greater than 1 percent contributes to the reported results).

5. In the MSLB accident (Calculation NE-02-04-06), an MSLB release can occur from a variety of locations. These locations appear to impact the volume of the steam released. Please explain how the analyses determined the worst case transport scenario.

6. The following questions pertain to the spray removal credit in containment and steam lines.

a) Per TS Bases B 3.6.2.2, "Suppression Pool Water Level," the suppression pool volume ranges between approximately 112,000 cubic feet at the low water level limit, of 30 feet, 9.75 inches, and approximately 114,000 cubic feet at the high water level limit, of 31 feet, 1.75 inches.

Chapter 15, Table 15.8-1, states that the minimum suppression pool liquid volume is 112,197 cubic feet. It also states that the water in the pedestal and water lower than 12 feet below the vent exit are not included in this volume.

The minimum volume of the suppression pool assumed in the LOCA calculation (NE-02-04-05) is 137,262 cubic feet. This value includes the pedestal and water 12 feet below the vent exits. Please justify the use of this volume and justify that the water in the pedestal and 12 feet below the vent exits can, and will, mix completely with the volume credited for determining the spray decontamination factor (DF) and for the determination of the doses from emergency core cooling system leakage.

b) Page 5.12 of NE-02-04-05 states:

No maximum DF is established for aerosol removal (as permitted by Reference 3), and there is no practical need to limit elemental iodine removal (since Revision I of

Reference 13 establishes a minimum elemental iodine partition coefficient, H, of 300 as long as the pH is greater than approximately 7.3).

Please specify where Reference 13 provides the partition coefficient for sodium pentaborate at a pH of 7.3 and justify its use, if it is not explicitly specified.

c)   Page 5.12 of NE-02-04-05 states:

The limited amount of elemental iodine initially present (4.85%) means that once the DF is applied, the percentage of elemental iodine remaining airborne would be approximately 0.04% of the total release. This is only 27% of the organic iodine percentage; and therefore, this amount may be neglected (particularly because the pH reaches 7.3 only after 30 days when the dominant dose contributor 1-131 has already been through 3.7 halflives).

Since this analysis focuses on the only the source contribution and does not consider the entire pathway for the releases, justify this argument considering the total impact to the dose.

d)   Please justify how elemental and particulate iodine removal in the drywell can be removed with the same removal coefficients.

e)   The staff would like to understand the methods used to determine the deposition velocity and flow rates used for aerosol, elemental, and organic deposition in the steamlines. Please provide LOCA dose analysis Reference 15 entitled, "Aerosol Removal in the Drywell and Steamlines," Document No. PSAT 206.QA.01.06.

f)   Please confirm that the surface areas used for the aerosol removal in the steamlines includes only horizontal piping and is calculated using the following equation:

$$Area = Diameter_{Internal} \times Horizontal\ Length$$

g)   Please confirm that the surface areas used for elemental iodine in the steamlines are calculated using the following equation:

$$Area = Diameter_{Internal} \times Length \times \pi$$

h)   Calculation NE-02-04-05, determines the flow rates of the leakage into steam lines. The volumetric flow rate is determined for both the intact and failed lines. Please describe the models, methods, assumptions, and justification for the assumptions and models used to calculate these flow rates.

7.   CGS proposes to no longer credit the main steam isolation valve leakage control system and remove the associated operability requirements from the TSs. How is potential

leakage from this system accounted for in the LOCA dose model and controlled by the TSs?

8. CGS has requested that credit for the standby gas treatment system (SGTS) is delayed for the first 20 minutes while a negative pressure condition is being established in secondary containment. The basis for this 20 minute drawdown needs to be clarified. Does the 20 minute drawdown include the time for the SGTS to become operational (time for an initiation signal, etc.)?

9. For the calculation of the aerosol removal rates in the drywell, CGS uses guidance from NUREG-0800, Section 6.5.2. To calculate the spray removal rate for particulate iodine the sprayed volume is used. Please justify the volume used.

10. Attachment 1, page 47, states ESF, "Leakage was assumed to start at t =15 minutes after the event." Please justify this assumption.

11. Attachment 2, page 9, states that CGS conforms to Regulatory Position 5.1.3. Regulatory Position 5.1.3 states:

> The numeric values that are chosen as inputs to the analyses required by Part 50.67 of 10 CFR [Title 10 of the *Code of Federal Regulations*] should be selected with the objective of determining a conservative postulated dose . . .

The ranges of flow rates out of the standby gas treatment system are given in the proposed TS 5.5 as 4320 to 5280 cubic feet per minute (cfm) for the SGTS and 900 to 1100 cfm for the control room emergency filtration (CREF) system.

a) The modeling of the SGTS appears to use a flow rate of 5000 cfm for the SGTS. Please justify why the nominal value is conservative when calculating a postulated dose.

b) The volumetric flow rate provided in Calculation NE-02-04-1 (Revision 1), Section 3.10, page 5.003, states the actual flow rate is 5378 ± 433.5 cfm. Please clarify the actual range of flows for this system.

12. Attachment 2, page 4, states that the nuclides used for CGS are:

> . . . the 60 identified as potentially important contributors [to total effective dose equivalent] in NUREG/CR-4691 (MACCS Users Guide) [less than two cobalt isotopes which have a minor impact] plus four additional noble gas isotopes from TID-14844, plus three other short-lived noble gas isotopes, plus Ba137m for a total of 66.

The staff believes the list from NUREG/CR-4691 is the same list as the default list provided for the RADTRAD computer code. The 60 radionuclides that are contained in the RADTRAD code were selected based upon a study that determined that those

60 radionuclides have the greatest impact on offsite dose assuming that each individual element has an equal release fraction.

    a)   Attachment 2, page 7, Section 4.2.3, states that the models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. It states that the CGS analysis conforms to this guidance. Confirm that the most conservative radionuclides were used to determine the source for the CGS shielding studies for the shine doses from external sources to the control room.

    b)   Please justify the changes to the default RADTRAD list of nuclides.

13.   Updated Final Safety Analysis Report (UFSAR) Section 4.1.2.1.3, "Fuel Assembly Description," states:

        The core is loaded with FANP [Framatome ANP] and Westinghouse Electric Company reload fuel. The Westinghouse Electric Company reload fuel assemblies are composed of a 10 x 10 array of fuel rods with a central, cruciform water channel (see Reference 4.1-8). The FANP reload fuel assemblies are composed of 10 x 10 array of fuel rods with a single, large, central water channel (Reference 4.1-20).

    Page 62 of Attachment 1 states that the fuel handling accident (FHA) analysis is based on an 8 x 8 fuel pin array with 250 fuel pins that are postulated to break. On page 18 of Attachment 2, CGS states that they conform to Regulatory Position 1.1, Appendix B Regulatory Guide 1.183. This Regulatory Position states, "[t]he number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case." For the allowed fuel designs in the CGS core please describe why the proposed analysis, based upon an 8x8 assembly, provides the limiting radiological consequences for the fuel handling accident.

14.   Page 63 of 91 of Attachment 1 states that:

        Based on the comparable water depth available for decontamination and the difference in the postulated drop distances, Energy Northwest concludes that the consequences of an FHA [fuel handling accident] over the reactor cavity bound those for an FHA over the transfer area or over the spent fuel pool. This conclusion is consistent with the [U.S. Nuclear Regulatory Commission (NRC)] staff conclusion for a similar configuration at the Fitzpatrick plant as documented in a recent Safety Evaluation Report (SER) (Reference 32).

The staff reviewed the Fitzpatrick SER and found that the basis of the staff's conclusion was as follows:

> ENO stated that the implied reduction in scrubbing efficiency is offset by the reduced number of fuel rods (i.e., 81 vs. 125) that are projected to be damaged by a fuel assembly drop over the spent fuel pool.

CGS has not provided a similar argument because CGS has not provided a value (and justifying analysis) for the reduced number of damaged fuel rods for an FHA in the fuel transfer area or over the spent fuel pool. If CGS plans to use this method of analysis please provide the number of fuel rods and the analysis justifying this number.

15. Page 62 of Attachment 1 states, "The TS minimum required water depth available over the point of fuel assembly impact is approximately 22 [feet], just 1 [foot] lower than the 23 [feet] upon which a DF of 200 is based." The discussion is focused on the water level above the point of impact for an FHA over the fuel transfer area or the spent fuel pool. It compares this depth of water to the 23 feet upon which the DF of 200 is based. The depth should be compared to a conservative release point of the radioactivity from the damaged fuel assemblies. Please provide details about the assumptions used to determine water depth above these release points. Please justify your assumptions. If the assembly is assumed to lie flat on top of the racks, justify why it is not possible that the fuel assembly could be in any other position.

16. Page 19 of 22 of Attachment 2 provides Table 3, "Comparison with Regulatory Guide 1.183, Appendix B." In Table 3, CGS states that the application conforms with Regulatory Position 5.3. Regulatory Position 5.3 states:

> If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.
>
> Footnote 3:
> The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

Several proposed TS changes (for example TS 3.6.4.1, 3.6.4.2, 3.6.4.3) delete the requirement for these systems to be operable and do not include any controls in the TS for manual isolation, if a fuel handling accident were to occur. Explain how the proposed TS and bases provide assurance that the intent of closure as a defense-in-depth measure is accomplished and that the open penetrations are closed.

17   General Design Criterion (GDC) 61 and 64 are part of the CGS licensing bases. For the proposed TSs changes describe how these criteria continue to be met for these proposed changes.

18.   Several proposed specification changes delete the Note: Limiting Condition for Operation (LCO) 3.0.3 is not applicable. For an example, see the required action for TS LCO 3.7.4, Action E. This change appears to deviate from the TSTF-51 traveler. Please provide a justification for this change.

19.   Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of a design. Generic Letter (GL) 2003-01, "Control Room Habitability," addresses issues with respect to assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing.

       Section 4.2 of Attachment 1 provides the test conditions and results of inleakage testing. This section does not provide a description of the inleakage testing performed in the normal operating mode credited in the FHA and control rod drop accident (CRDA). In light of your Appendix B requirements, GL 2003-01, and because the 1100 cfm value for unfiltered inleakage is not based upon a measurement during this mode of operation, justification should be provided to explain why this number is appropriate. Please provide information as to how CGS has confirmed the inleakage characteristics of the control room envelope in the normal operating mode credited for the duration of the FHA and CRDA. Please provide details regarding your control room, design, maintenance, and assessments to justify the use of and any plans to verify this number.

       e)   Does the 1100 cfm unfiltered inleakage include 10 cfm for ingress and egress into and out of the control room over the duration of the accident?

       f)   How much of the 1100 cfm is due to forced design flow and how much is assumed for unfiltered inleakage due to inflow other than egress and ingress?

       g)   The use of 1100 cfm of unfiltered inleakage for the design bases FHA and CRDA meets the requirements of 10 CFR 50.36, Criterion 2. This value is "a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier." Since use of the 1100 cfm unfiltered inleakage value meets 50.36, Criterion 2, and the system which supplies this inflow essentially replaces the TS for the control room emergency filtration (CREF) system (TS 3.7.3), please justify why CGS has not proposed a limiting condition for operation for this flow rate or provide a limiting condition for operation for this value.

20.   Regulatory Position 5.1.2 of RG 1.183 states:

>   Credit may be taken for accident mitigation features that are
>   classified as safety-related, are required to be operable by
>   technical specifications, are powered by emergency power
>   sources, and are either automatically actuated or, in limited cases,
>   have actuation requirements explicitly addressed in emergency
>   operating procedures. The single active component failure that
>   results in the most limiting radiological consequences should be
>   assumed. Assumptions regarding the occurrence and timing of a
>   loss of offsite power should be selected with the objective of
>   maximizing the postulated radiological consequences.

Attachment 2, Table 1, states that the CGS analysis conforms to this regulatory position. For the FHA and CRDA accident, please provide additional details how CGS control room heating, ventilation, and air conditioning credited for these accidents conforms to Regulatory Position 5.1.2. State whether the system operation credited is operable by the TS, is powered by emergency power sources, credits the worst case single failure and models the occurrence and timing of a loss-of-offsite power. Please provide justification for these answers.

21.   Page 19 of Attachment 2 states that CGS conforms to Regulatory Position 5.3. Regulatory Position 5.3 states:

>   If the containment is open during fuel handling operations (e.g.,
>   personnel air lock or equipment hatch is open),[3] the radioactive
>   material that escapes from the reactor cavity pool to the
>   containment is released to the environment over a 2-hour time
>   period.

Page 67, Section 4.7.4 states:

>   For modeling purposes, a fractional release rate of 2.3 volumes
>   per hours was utilized to ensure that at least 99% of the activity
>   was released from the reactor building during the first 2 hours.

If less than 100 percent is released from the building how does this conform to Regulatory Position 5.3?

22.   CGS proposes to remove the following words from page B.3.6.4.1-3 of the TSs, "CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable."

Attachment 1, page 62 states:

>   The analysis assumed a ground level release from the reactor
>   building over a 2-hour period. No credit was taken for secondary
>   containment, the [SGTS] or the CREF system. The assumptions
>   used in this analysis are consistent with RG 1.183.

The analysis that supports the removal of this statement assumes a certain containment configuration and resulting leakage pathways out of containment. Please confirm that this configuration provides the most bounding atmospheric dispersion factors for all possible release paths allowed by removing this statement. For example, if this statement is removed there would appear to be no controls on the configuration of containment. It might be possible that hatches or scuttles could be opened that would lead to a more direct leakage pathway to the control room. Please verify that your analysis bounds all possible containment configurations.

23.   Page 5.0 of Calculation NE-02-04-08 states that the "X/Q values take into account that [the] release [of radioactivity] begins 24 hours after the accident." The accident is typically assumed to begin when the fuel assembly is dropped and not at the time of shutdown. Please justify using an X/Q that models the accident starting at the shutdown of the reactor.

24.   Please justify the removal of the fuel handling accident reference from the bases of TS 3.3.6.2 (page B 3.3.6.2-12, "Secondary Containment Isolation Instrumentations.").

25.   Page 82 of the submittal states:

> Based on the overall reduction in [control room] operator dose due to AST methodology, similarities in ventilation systems, and the ability to evacuate the TSC [technical support center], an updated quantitative assessment of the TSC dose based on the AST source term was not performed.

Likewise, the emergency operations facility (EOF) doses were not reassessed based upon a quantitative assessment of the doses. The NRC staff requests further information regarding the quantitative assessment of the TSC and EOF doses.

a)   Does Columbia currently have in its licensing bases commitments to meet GDC 19 for the TSC and EOF without compensatory actions such as evacuations?

b)   Please provide enough detail regarding the qualitative assessment for the NRC staff to come to the same conclusion as the Columbia evaluation. Consider that the reduction in doses due the AST may be due to factors that are independent of the TSC and EOF doses (such as atmospheric dispersion factors etc. before isolation).

26.   Page 5.007 of NE-02-04-06 states that consistency with the current licensing basis is maintained by the position that only the reactor coolant liquid contains the iodine. UFSAR, Section 15.6.4.5 states:

> The only activity available for release from the break is that which is present in the reactor coolant and steam lines prior to the break. The iodine inventories and the subsequent exposures are based

on the equilibrium conditions and maximum reactor coolant activity for an iodine spiking event as allowed by the Technical Specifications.

Iodine partitions into the steam in the steam lines. Justify why the iodine in this steam is not included in the source term used to calculate the doses in NE-02-04-06.

27. Attachment 1, page 14, states, "This change does not affect any accident analysis and does not affect the operation of the plant during refueling activities." Please justify why this does not affect any accident analyses if this change establishes an operational requirement that is consistent with the assumptions in the AST FHA analysis.

28. Regulatory Position 6.2, Appendix A, Regulatory Guide 1.183 states:

> All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.

Please justify the 50 percent reduction assumed, and provide the supporting site-specific analyses.

29. Did the onsite meteorological measurement program meet the recommendations of RG 1.23, "Onsite Meteorological Programs," from 1996-1999? If not, please describe the deviations and provide justification that the deviations did not significantly impact collection of high quality data.

30. A detailed examination of the 1996-1999 onsite meteorological data revealed a few irregularities. For example, beginning in mid-June 1996, approximately two months of the daytime atmospheric stability measurements were shown as invalid while most of the nighttime measurements were provided as valid. At the end of December 1999, the hourly stability was listed as Category E for a period of approximately 250 consecutive hours. Staff made a year-by-year comparison of ARCON96-generated X/Q values using the 1996-1999 Columbia data and found larger year-to-year differences in the X/Q values than expected. Therefore, please provide:

a) A summary description of any significant changes in the Columbia onsite meteorological measurement program since 1996, and

b) Hourly data for a representative period of record (e.g., three years) following any post-1999 changes in the measurement program. Please include a summary description of the data (e.g., height and units of measurement) for each parameter. The data may be provided as "raw" hourly electronic summaries (e.g., do not need to be in ARCON96 format) if the format and data description facilitate ready examination and conversion of the units of the data by a commercially available computer spreadsheet.

31. Do each of the three control room air intakes meet applicable design criteria of an ESF, including single-failure criterion, missile protection, seismic criteria, and operability under loss-of-offsite alternating current power conditions? The staff was unable to confirm that the local and remote-1 control room air intakes are in separate 90 degree windows for all of the postulated release locations evaluated using ARCON96. If two intakes could concurrently be in a single 90-degree window for a given release point, then a weighting factor, such as that described in Section 3.3.2.1 of RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," should be applied when calculating the effective X/Q values. Therefore, please provide a figure drawn to scale demonstrating that no two intakes are within the same 90-degree window for any of the postulated release points.

32. The effective atmospheric dispersion factors (X/Q values) shown in Tables 3 through 6 (pages 5.5 through 5.8) of Calculation No. NE-02-03-14 (attachment to the September 30, 2004, submittal) are based on the assumption that the local, remote-1 and remote-2, intakes are concurrently drawing air throughout the course of each accident. Flow into the two remote intakes is assumed to be the same at any point in time but can change as a function of time. Although closed, the local intake is assumed to draw 150 cubic feet per minute of filtered flow under all scenarios. The calculations assume that the intake with the largest associated X/Q value is drawing the contaminated air and that the contamination is reduced by dilution with clean air from the other two intakes.

   a) Please confirm that closure of a remote intake as discussed in the submittal (e.g., page 5.3 of Calculation No. NE-02-03-14 item C) would not result in higher effective X/Q values than those listed in Tables 3 through 6.

   b) Item 2 on page 5.3 of Calculation No. NE-02-03-14 states that the filtered flow rate for the closed local intake is assumed to be 150 cubic feet per second. Have tests or other procedures been performed to confirm this value? If the flow could be significantly more or less than 150 cubic feet per second, then the effective X/Q values listed in Tables 3 through 6 may not be the limiting values.

   c) What provisions are in place to assess and potentially revise the effective X/Q values in the event that the flow rates could change to an unanalyzed or more limiting condition?

33. Attachment 1 to Calculation No. NE-02-03-14 provides X/Q values for an assumed release from the CST to the remote-1 intake. It is noted that the CST is closest to this intake. What are the approximate distances of the CST to the remote-2 and local intakes to justify the assumption that a release to the remote-1 intake would be limiting? Could more than one intake be in the same 90-degree window for this postulated release?

34. Could more than one release scenario occur for any of the design-basis accidents addressed in this license amendment request? For example, could effluent be released to the environment from a different location should loss-of-offsite power or other single failure occur?

Columbia Generating Station

November 2005