



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 251
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Tennessee Valley Authority (the licensee), dated December 11, 1996, as supplemented by letter dated November 3, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

9805270018 980518
PDR ADOCK 05000260
PDR



11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100

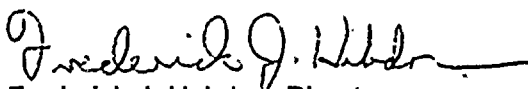
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 251 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

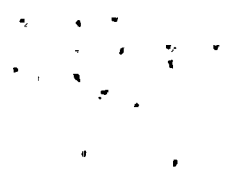
FOR THE NUCLEAR REGULATORY COMMISSION



Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: May 18, 1998



ATTACHMENT TO LICENSE AMENDMENT NO. 251

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. An overleaf page is included to maintain document completeness.

Remove

1.2/2.2-1

1.2/2.2-2

3.6/4.6-30

3.6/4.6-31

Insert

1.2/2.2-1

1.2/2.2-2

3.6/4.6-30

3.6/4.6-31 *

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

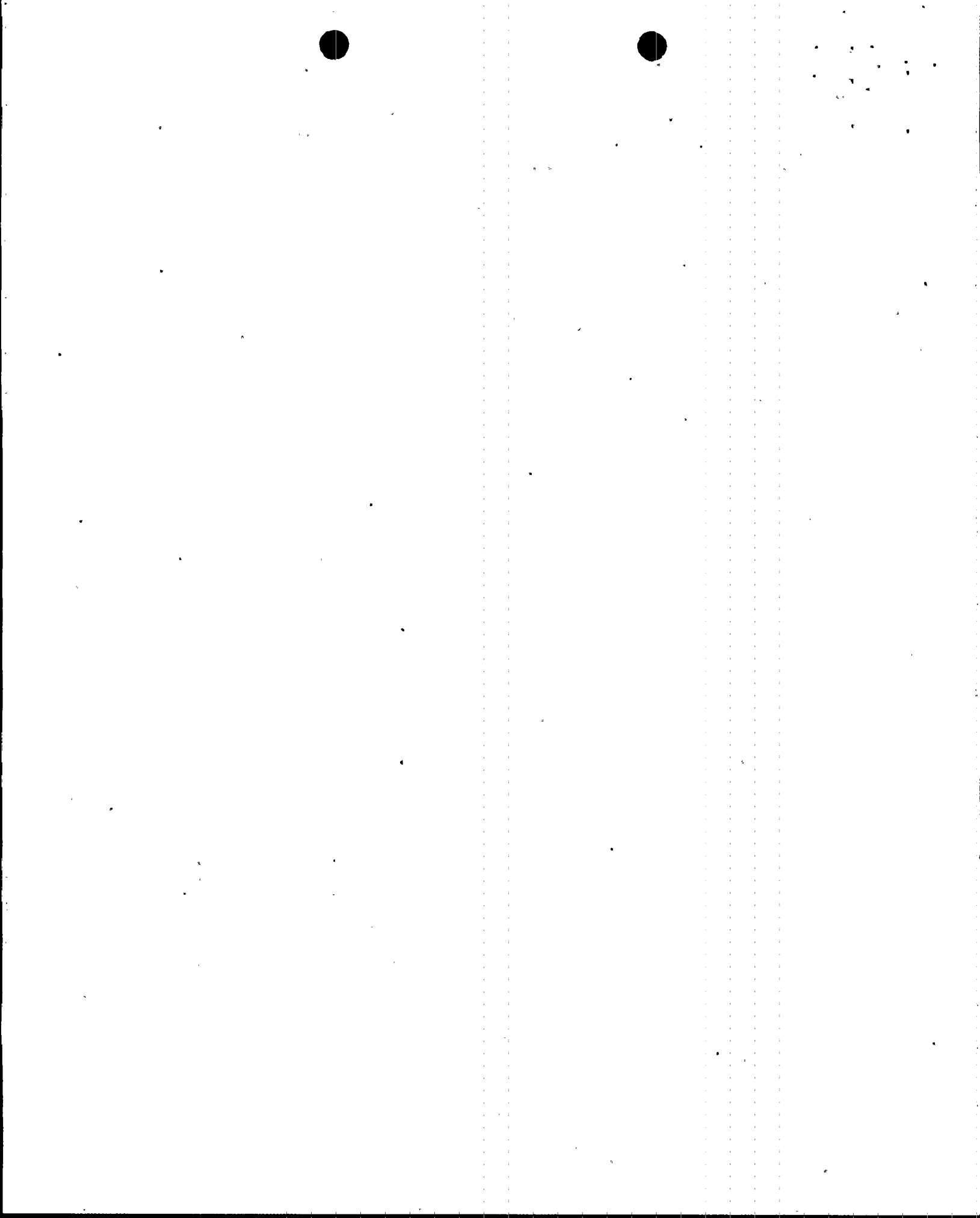
- A. Verify the safety function lift settings of the required S/RVs are within \pm three percent of the setpoint as follows:

<u>Number of S/RVs</u>	<u>Setpoint (psig)</u>
4	1105
4	1115
5	1125

Following testing, lift settings shall be within \pm one percent.

Limiting Safety
Protective Action System Setting

- B. Scram--nuclear $\leq 1,055$ psig system high pressure



1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are not high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing report for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.4 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the



Small, illegible markings or text in the top right corner.

3.6/4.6 BASES

3.6.B/4.6.C (Cont'd)

five gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increases over any 24-hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

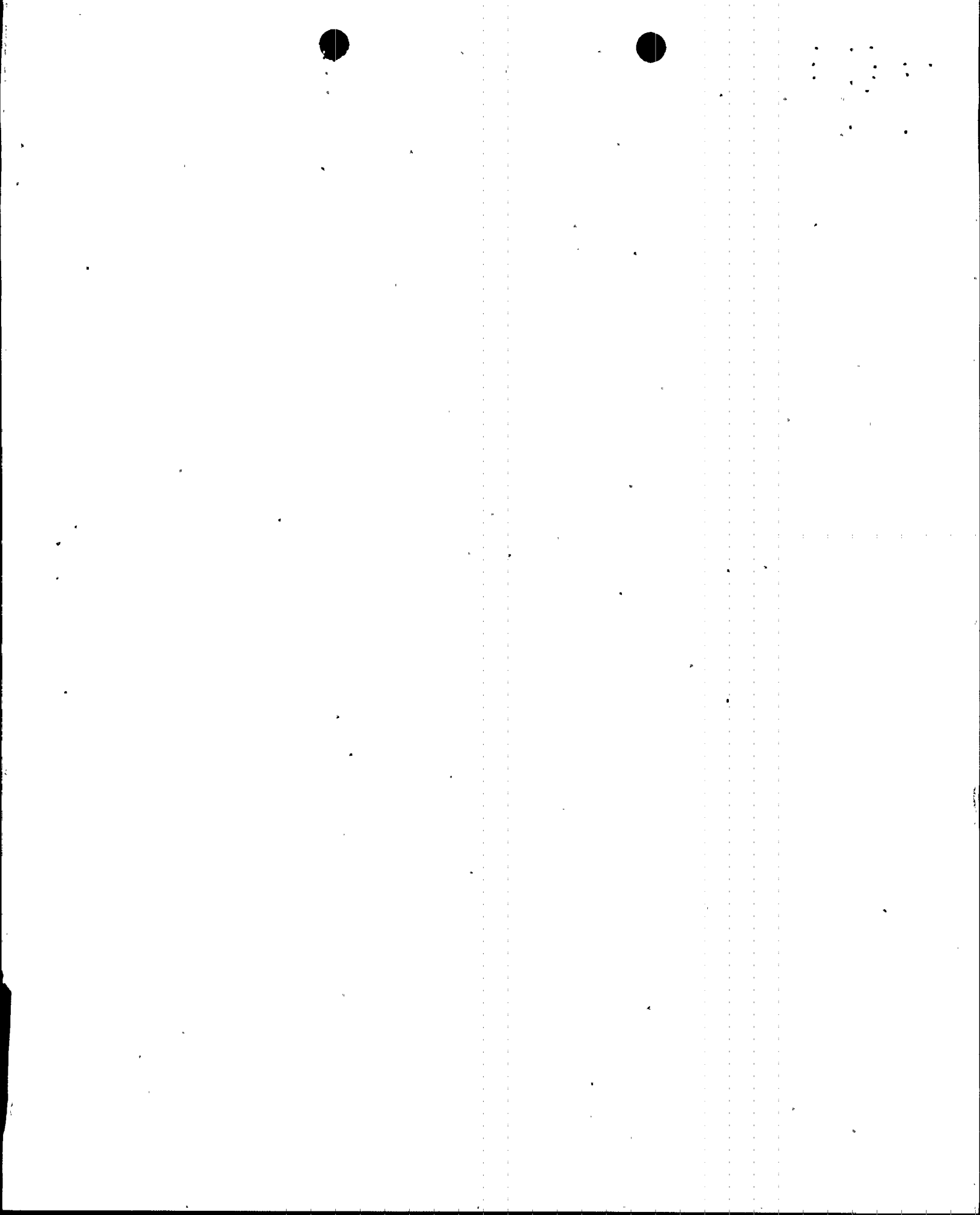
1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per cycle is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within their specified tolerances. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.



3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

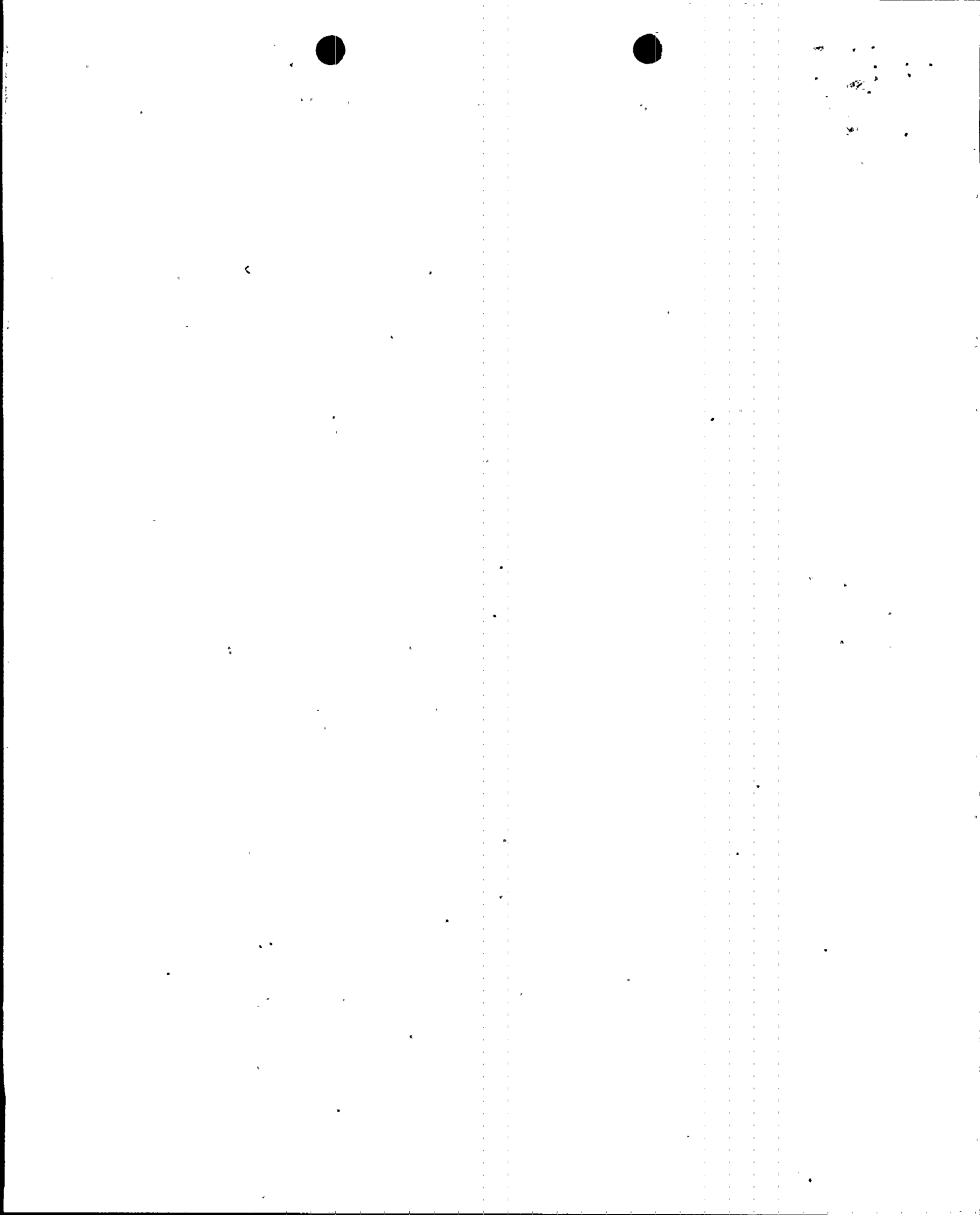
REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973
4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.





UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

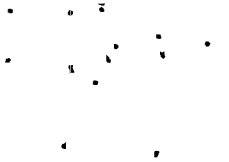
DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 210
License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Tennessee Valley Authority (the licensee) dated December 11, 1996, as supplemented by letter dated November 3, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.



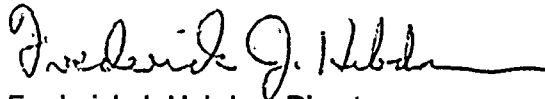
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 210, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: May 18, 1998



ATTACHMENT TO LICENSE AMENDMENT NO.210

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. An overleaf page is included to maintain document completeness.

Remove

1.2/2.2-1

1.2/2.2-2

3.6/4.6-30

3.6/4.6-31

Insert

1.2/2.2-1

1.2/2.2-2

3.6/4.6-30

3.6/4.6-31 *



• • •
• • •
• • •

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

- A. Verify the safety function lift settings of the required S/RVs are within \pm three percent of the setpoint as follows:

<u>Number of S/RVs</u>	<u>Setpoint (psig)</u>
4	1105
4	1115
5	1125

Following testing, lift settings shall be within one percent.

Limiting Safety Protective Action System Setting

- B. Scram--nuclear \leq 1,055 psig system high pressure



1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

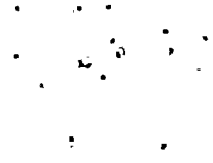
The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing report for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.4 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the



3.6/4.6 BASES

3.6.C/4.6.C (Cont'd)

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increases over any 24-hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

References

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per cycle is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within their specified tolerances. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.



3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

References

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973
4. Generic Reload Fuel Application, Licensing Topical Report, NEDE 24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly hold-down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is

