PROPOSED CHANGE NO. 66

COOPER NUCLEAR STATION TECHNICAL SPECIFICATIONS

PROPOSED PAGE CHANGES

SAFETY LIMIT

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.

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications:

1. The reactor vessel dome pressure shall not exceed 1337 psig at any time when irradiated fuel is present in the reactor vessel.

2. The reactor vessel dome pressure shall not exceed 75 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

LIMITING SAFETY SYSTEM SETTING

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:

1. The limiting safety system settings shall be as specified below:

   Protective Action/Limiting Safety System Setting

   A. Scram on Reactor Vessel high pressure-

      ≤1045 psig

   B. Relief valve settings-

      ≤1210 psig

   C. Safety valve settings-

      ≤1277 psig

2. Action shall be taken to decrease the reactor vessel dome pressure below 75 psig or the shutdown cooling isolation valves shall be closed.

2.2 BASES

The eight relief valves and three safety valves are sized, and the maximum upper limit on the set pressures are established, in order to ensure that the peak vessel bottom pressure remains below 110% of the vessel design pressure, as required by Section III of the ASME code. The relief valve settings satisfy the code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. The peak vessel dome pressure limit of 1337 psig was established based upon the ASME limit of 1375 psig (110% of vessel design pressure) and the difference between the pressure at the bottom of the vessel head and that in the steam dome of the reactor vessel. The relief valve settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients.

Reanalysis in Reference 7 for the limiting vessel overpressure event, the MSIV-Closure with flux scram event, was performed with the setpoints of all the relief valves assumed to be at an upper limit of 1210 psig and all the safety valves at 1277 psig. The upper limit setpoint for the relief valves was established to ensure that the peak vessel pressure will remain below 1375 psig. The peak vessel bottom pressure resulting from the Reference 7 evaluation was 1322 psig, which provides sufficient margin to ensure that the 1375 psig (110% of the vessel design pressure) limit will not be exceeded. Results of the overpressure protection analysis for the current cycle are provided in the current reload licensing document. The actual relief valve and safety valve setpoints will be grouped as specified in Section 4.6.D. A sensitivity study on peak vessel pressure to the failure to open of one of the lowest set-point safety valves was performed for a typical high power density BWR (Reference 8). The study is applicable to the Cooper reactor and shows that the sensitivity of a high power density plant to the failure of a safety valve is approximately 20 psi. A plant specific analysis for the Cooper overpressure transient would show results equal to or less than this value.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

REFERENCES

1. Topical Report, "Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor", General Electric Company, Atomic Power Equipment Department (APED-5698)

2. Station Nuclear Safety Operational Analysis (Appendix G)

3. Station Safety Analysis (Section XIV)

4. Control and Instrumentation (Section VII)

5. Summary Technical Report of Reactor Vessel Overpressure Protection (Question 4.20, Amendment 11 to SAR)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

7. "SRV Setpoint Tolerance Analysis for Cooper Nuclear Station," General Electric Company, NEDC-31628P, October 1988.

8. Letter from I. F. Stuart (GE) to V. Stello (NRC) dated December 23, 1975.

3.6.D Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, all three safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.

2.

a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.

b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.

3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to a cold shutdown condition within 24 hours.

4. From and after the date that position indication on any one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve position indication is sooner made operable.

4.6.D. Safety and Relief Valves

1. Approximately half of the safety valves and relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

The safety valve function of safety and relief valves shall actuate with the following settings:

a. Relief Valves

1080 psig ± 33 psi (2 valves)
1090 psig ± 33 psi (3 valves)
1100 psig ± 33 psi (3 valves)

b. Safety Valves

1240 psig ± 37 psi (3 valves)

2. At least one of the relief valves shall be disassembled and inspected each refueling outage.

3. Deleted

4. Deleted

5. Once per operating cycle, with the reactor pressure ≥ 100 psig, each relief valve shall be manually opened until the main turbine bypass valves have closed to compensate for relief valve opening.

6.

a. Operability of the relief valve position indicating pressure switches and the safety valve position indicating thermocouples shall be demonstrated once per operating cycle.

b. An Instrument Check of the safety and relief valve position indicating devices shall be performed monthly.

indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks, associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 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 plant should be shutdown to allow further investigation and corrective action.

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

Reactor coolant leakage is also sensed by the containment radiation monitoring unit which senses gross beta, gamma particulate and iodine as well as by oxygen and hydrogen analyzers. Leakage can also be detected by area temperature detectors, humidity detectors and pressure instrumentation. Due to the many and varied ways of detecting primary leakage, a 30 day allowable repair time is justified.

D. Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (113 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for Cooper Nuclear Station has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protective criteria of the ASME code. Refer to Specification 2.2 and the corresponding bases for the maximum opening upper limit pressure which has been established for Cooper Nuclear Station. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis IV.4.2.1 of subsection IV.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the complete safety analysis, which includes demonstration of compliance with the ASME code requirements for vessel overprotection, is presented in Reference 1. Results of the overpressure protection analysis for the current cycle are provided in the current reload licensing document.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations.

The relief and safety valves are bench tested every second operating cycle to ensure their set points are within the ±3 percent tolerance. To avoid increasing the probability of setpoint drift, all valves are to be reset to within ±1% of their nominal setpoint prior to being returned to service. Additionally, once per operating cycle, each relief valve is tested manually with reactor pressure above 100 psig to demonstrate its ability to pass steam.

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 starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The position indicating pressure switches for the relief valves and the thermocouples for the safety valves serve as a diagnostic aid to the operator in the event of a safety/relief valve failure. If position indication is lost, alternate means are available to the operator to determine if a safety valve is leaking.

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 bases double-ended line break. Therefore, if a failure occurs, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5%, 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%, riser and nozzle assembly integrity has been verified. If they do differ by 10% 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 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a

BASES (cont'd)

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

In cases where the cause of failure has been identified, additional snubbers, having a high probability for the same type of failure or that are being used in the same application that caused the failure, shall be tested. This requirement increases the probability of locating inoperable snubbers without testing 100% of the snubbers.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. Ten percent of each type of snubber represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers in high radiation areas or those especially difficult to remove need not be selected for functional tests provided operability was previously verified.

The service life of a snubber is evaluated via manufacturer input and consideration of the snubber service conditions. The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

REFERENCES

1. "SRV Setpoint Tolerance Analysis for Cooper Nuclear Station," General Electric Company, NEDC-31628P, October 1988.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, David J. Robare, being duly sworn, depose and state as follows:

1. I am Manager, Licensing Services, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. The information sought to be withheld is contained in the GE proprietary report NEDC-31628P (Class III), "SRV Setpoint Tolerance Analysis for Cooper Nuclear Station", October 1988. This report provides the basis for revised plant Technical Specification changes to SRVs and SVs consistent with current ASME Code provisions. The safety limit setting tolerance in Section 2.2 and the surveillance requirements of Section 4.6.D of the Tech Specs could be revised. In addition, it provides recommended setpoint tolerances to be applied prior to returning valves to service.

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it. A substantial element of secrecy must exist so that, except by the use of improper means, there would be difficulty in acquiring information...Some factors to be considered in determining whether given information is one's trade secret are (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expanded by him developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others".

3. Some examples of categories of information which fit into the definition of Proprietary Information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

   c. Information which if used by a competitor, would reduce his expenditures of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

4. Initial approval of proprietary treatment of a document is typically made by the Subsection Manager of the originating component, the person who is most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents are clearly identified as proprietary.

5. The procedure for approval of external release of such a document typically requires review by the Subsection Manager, Project Manager, Principal Scientist or other equivalent authority, by the Subsection Manager of the cognizant Marketing function (or delegate) and by the Legal Operation for technical content, competitively effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees then only with appropriate protection by applicable regulatory provisions or proprietary agreements.

6. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

7. The information to the best of my knowledge and belief has consistently been held in confidence by the General Electric Company, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties have been made pursuant to regulatory provisions of proprietary agreements which provide for maintenance of the information in confidence.

8. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company and deprive or reduce the availability of profit making opportunities because it would provide other parties, including competitors, with valuable information.

GENERAL ELECTRIC COMPANY

STATE OF CALIFORNIA }
} **ss:**
COUNTY OF SANTA CLARA }

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are truly and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 6 day of OCTOBER 19 89.

David J. Robare
David J. Robare
General Electric Company

Subscribed and sworn before me this 6th day of October 19 89.

OFFICIAL SEAL
MARY L KENDALL
Notary Public-California
SANTA CLARA COUNTY
My Comm. Exp. Mar. 26, 1993

Mary L. Kendall
Notary Public, State of California