PDR

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-364

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-8 issued to Southern Nuclear Operating Company, Inc. (the licensee), for operation of the Joseph M. Farley Nuclear Plant, (Farley) Unit 2, located in Houston County, Alabama.

The proposed amendment would modify the Technical Specifications (TS), on an interim basis, to allow the implementation of interim plugging criteria for tube support plate elevations. The amendment would also modify the TS to reduce the total allowable primary-to-secondary operational leakage from any one steam generator from 500 gallons per day to 150 gallons per day. The total allowable primary-to-secondary operational leakage through all steam generators will be reduced from one gallon per minute (1,440 gallons per day) to 450 gallons per day.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's

DF03 1/0

regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Operation of Farley Unit 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Testing of model boiler specimens for free standing tubes at room temperature conditions show burst pressures as high as 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 30 volts. Burst testing performed on pulled tubes with up to 10 volt indications show burst pressures in excess of 5900 psi at room temperature. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the [USNRC Regulatory Guide] R.G. 1.121 criterion requiring the maintenance of a margin of three times normal operating pressure differential on tube burst if through-wall cracks are present. Based on the existing data base, this criterion is satisfied with bobbin coil indications with signal amplitudes less than 6.2 volts, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data. The 1.0 threshold volt criteria provides an extremely conservative margin of safety to the structural limit considering expected growth rates of [outer diameter stress corrosion cracking] ODSCC at Farley. Alternate crack morphologies can correspond to 6.2 volts so that a unique crack length is not defined by a burst pressure to voltage correlation. However, relative to expected leakage during normal operating conditions, no field leakage has been reported from tubes with indications with a voltage level of under 7.7 volts (as compared to the 1.0 volt proposed interim tube plugging limit).

   Relative to the expected leakage during accident condition loading, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station

Auxiliaries, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. Of these, the Major Secondary System Pipe Failure is the most limiting for Farley Unit 2 in considering the potential for off-site doses. The offsite dose analyses for the other events which model primary-to-secondary leakage and steam release from the secondary side to the environment assume that the secondary side remains intact. The steam generator tubes are not subjected to a sustained increase in differential pressure, as is the case following a steam line break [SLB] event. This increase in differential pressure is responsible for the postulated increase in leakage and associated offsite doses following a steam line break event. Upon implementation of the interim plugging criteria, it must be verified that the expected distribution of cracking indications at the tube support plate intersections are such that primary-to-secondary leakage would result in site boundary dose within the current licensing basis for Unit 2, 1 gallon per minute during a steam line break event. Data indicate that a threshold voltage of 2.8 volts would result in through-wall cracks long enough to leak at SLB conditions. Application of the proposed plugging criteria requires that the current distribution of a number of indications versus voltage be obtained during the Unit 2 Eighth Refueling Outage. The current voltage is then combined with the rate of change in voltage measurements to establish an end of cycle voltage distribution and, thus, leak rate during SLB pressure differential. If it is found that the potential SLB leakage for degraded intersections planned to be left in service exceeds 1 gallon per minute, then additional tubes will be plugged or repaired to reduce SLB leakage potential to 1 gallon per minute or less.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Implementation of the proposed interim tube support plate elevation steam generator tube plugging criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions). The bobbin probe signal amplitude plugging criteria is established such that operational leakage or excessive leakage during a postulated steam line break condition is not anticipated.

   [Southern Nuclear Operating Company, Inc.] SNC will implement a maximum leakage rate limit of 150 [gallons per day] gpd per steam generator to help preclude the potential for excessive leakage during all plant conditions upon application of the plugging criteria. The R.G. 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of three against bursting a normal operating pressure differential. A voltage amplitude of 6.2 volts for typical OD SCC corresponds to meeting this tube burst requirement at the lower 95% uncertainty limit on the burst correlation. Alternate crack morphologies can correspond to 6.2 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at three times normal operating pressure differential and SLB conditions are about 0.42 inch and 0.84 inch, respectively. Normal leakage for these crack lengths would range from 0.11 gallons per minute to 4.5 gallons per minute, respectively, while lower 95% confidence level leak rates would range from about 0.02 gallons per minute to 0.6 gallons per minute, respectively.

An operating leak rate of 150 gpd will be implemented in application of the tube plugging limit. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the lower 95% confidence level leak rates. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

3. The proposed license amendment does not involve a significant reduction in margin of safety.

The use of the interim tube support plate elevation plugging criteria at Farley Unit 2 is demonstrated to maintain steam generator tube integrity commensurate with the requirements of R.G. 1.121. R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of OD SCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria is applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during all plant conditions and, hence, help to demonstrate radiological conditions are less than a small fraction of the 10 CFR [Part] 100 guideline.

In addressing the combined effects of [loss-of-coolant accident] LOCA + [safe shutdown earthquake] SSE on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the [reactor coolant system] RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential [that partial] through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley Unit [2] reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley Unit 2 and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley Unit 2, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on recent analyses results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that currently allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube plugging criteria is supplemented by 100% inspection requirements at the tube support plate elevations having OD SCC indications, reduced operating leak rate limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating pancake coil inspection requirements for the larger indications left in service to characterize the principal degradation mechanism as OD SCC.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs would reduce the RCS flow margin, thus implementation of the interim plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

The NRC staff has reviewed the licensee's analysis. In addition, with respect to the third standard, the NRC staff has considered the potential for a reduction in the margin to burst for tubes using the proposed criteria and finds that the margin to burst is comparable to that provided by the existing Technical Specification requirements. Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within thirty (30) days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room P-223, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland, from 7:30 a.m. to 4:15 p.m. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By MAR 30 1992, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR

2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at Houston-Love Memorial Library, 212 W. Burdeshaw Street, P. O. Box 1369, Dothan, Alabama. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board Panel, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board Panel will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman

Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 325-6000 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Elinor G. Adensam: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to James H. Miller, III, Esq., Balch and Bingham, P. O. Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board Panel that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated February 20, 1992, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW.

Washington, DC 20555 and at the local public document room located at Houston-Love Memorial Library, 212 W. Burdeshaw Street, P. O. Box 1369, Dothan, Alabama 36302.

Dated at Rockville, Maryland, this 26th day of February 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen T. Hoffman

Stephen T. Hoffman, Project Manager
Project Directorate II-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation