

DEC 31 1981

Docket No.: STN 50-482

Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
Wichita, Kansas 67201

Dear Mr. Koester:

Subject: Request for Additional Information for the Review of the
Wolf Creek Plant, Unit 1 Regarding Reactor Systems

As a result of our continuing review of the Wolf Creek Plant, Unit 1 FSAR, we find that we need additional information to complete our evaluation. The specific information required is in the area of reactor systems and is presented in the Enclosure.

To maintain our licensing review schedule for the Wolf Creek Plant FSAR, we will need responses to the enclosed request by February 4, 1982. If you cannot meet this date, please inform us within seven days after receipt of this letter of the date you plan to submit your responses so that we may review our schedule for any necessary changes.

Please contact Jon Hopkins, Wolf Creek Licensing Project Manager, if you desire any discussion or clarification of the enclosed request.

Sincerely,

Original signed by:
B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
As stated

cc: See next page

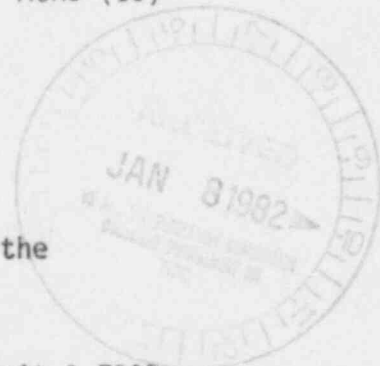
The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.*

OFFICE	DL:LB#1	DL:LB#1					
SURNAME	JHopkins/yt	BJYoungblood					
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TERA
NRC/PDR
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NSIC
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ACRS (16)



Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
Post Office Box 208
Wichita, Kansas 67201

cc: Mr. Nicholas A. Petrick
Executive Director, SNUPPS
5 Choke Cherry Road
Rockville, Maryland 20750

Mr. Jay Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Donald T. McPhee
Vice President - Production
Kansas City Power and Light Company
1330 Baltimore Avenue
Post Office Box 679
Kansas City, Missouri 64141

Ms. Mary Ellen Salva
Route 1, Box 56
Burlington, Kansas 66839

A. Scott Cauger, Esq.
Assistant General Counsel
Public Service Commission
P. O. Box 360
Jefferson City, Missouri 65102

Mr. Tom Vandel
Resident Inspector/Wolf Creek NPS
c/o U.S.N.R.C.
Post Office Box 311
Burlington, Kansas 66839

Mr. Michael C. Kenner
Wolf Creek Project Director
State Corporation Commission
State of Kansas
Fourth Floor, State Office Bldg.
Topeka, Kansas 66612

Ms. Wanda Christy
515 N. 1st Street
Burlington, Kansas 66839

Eric A. Eisen, Esq.
Birch, Horton, Bittner & Moore
1140 Connecticut Avenue, N. W.
Washington, D. C. 20036

Kansans for Sensible Energy
Post Office Box 3192
Wichita, Kansas 67201

REQUEST FOR ADDITIONAL INFORMATION

LOCKED REACTOR COOLANT PUMP ANALYSIS

440.0WC Reactor Systems Branch

440.3WC The analyses of a locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3 and GDC 17, we require that this event be analyzed assuming turbine trip and coincident loss of offsite power to the undamaged pumps.

Appropriate delay times may be assumed for loss of offsite power if suitably justified.

Steam generator tube leakage should be assumed at the rates specified in the Technical Specifications.

The event should also be analyzed assuming the worst single failure of a safety-system active component. Maximum technical specification primary system activity and steam generator tube leakage should be assumed. The analyses should demonstrate that offsite doses are less than the 10 CFR 100 guidelines values.